# SAFETY EVALUATION REPORT

# FOR THE STANDARDIZED NUHOMS® SYSTEM

# CERTIFICATE OF COMPLIANCE NO. 1004

# RENEWAL

# DOCKET NO. 72-1004

**Office of Nuclear Material Safety and Safeguards**
**United States Nuclear Regulatory Commission**
**May 2017**

# CONTENTS

Section Page

INTRODUCTION ..... v

1 GENERAL INFORMATION ..... 1-1
1.1 CoC and CoC Holder Information ..... 1-1
1.2 Safety Review ..... 1-1
1.3 Application Content ..... 1-1
1.4 Interim Staff Guidance ..... 1-2
1.5 Evaluation Findings ..... 1-2

2 SCOPING EVALUATION ..... 2-1
2.1 Scoping and Screening Methodology ..... 2-1
2.1.1 Scoping Process ..... 2-3
2.1.2 Structures, Systems, and Components Within Scope of Renewal ..... 2-7
2.1.3 Structures, Systems, and Components NOT Within Scope of Renewal ..... 2-17
2.2 Evaluation Findings ..... 2-20

3 AGING MANAGEMENT REVIEW ..... 3-1
3.1 Review Objective ..... 3-1
3.2 AMR Process ..... 3-1
3.3 Aging Management Review Results: Materials, Service Environment, Aging Effects, and Aging Management Programs ..... 3-2
3.3.1 Spent Fuel Assembly (SFA) ..... 3-14
3.3.1.1 Materials and Environments ..... 3-15
3.3.1.2 Aging Effects/Mechanisms for the Spent Nuclear Fuel Assemblies During the Period of Extended Storage ..... 3-19
3.3.1.3 Evaluation of the Proposed Aging Management Activities ..... 3-21
3.3.2 Dry Shielded Canisters (DSC) ..... 3-21
3.3.2.1 Materials and Environments ..... 3-22
3.3.2.2 Aging Effects/Mechanisms for the Dry Shielded Canisters During the Period of Extended Storage ..... 3-25
3.3.2.3 Evaluation of the Proposed Aging Management Activities ..... 3-35
3.3.3 Horizontal Storage Modules (HSM) ..... 3-37
3.3.3.1 Materials and Environments ..... 3-38
3.3.3.2 Aging Effects/Mechanisms for the Horizontal Storage Modules During the Period of Extended Storage ..... 3-41
3.3.3.2.1 Reinforced Concrete Structure ..... 3-41
3.3.3.2.2 Steel and Aluminum Subcomponents ..... 3-42
3.3.3.3 Evaluation of the Proposed Aging Management Activities ..... 3-48
3.3.4 Storage Pad (Basemat) ..... 3-48
3.3.4.1 Materials and Environments ..... 3-49
3.3.4.2 Aging Effects/Mechanisms for the Storage Pad (Basemat) During the Period of Extended Storage ..... 3-49
3.3.4.3 Evaluation of the Proposed Aging Management Activities ..... 3-49
3.3.5 Transfer Cask (TC) ..... 3-50

3.3.5.1 Material and Environments ..... 3-51
3.3.5.2 Aging Effects/Mechanisms for the Transfer Cask During the Period of Extended Storage ..... 3-52
3.3.5.3 Evaluation of the Proposed Aging Management Activities ..... 3-56
3.3.6 Evaluation Findings ..... 3-57
3.4 Time-Limited Aging Analyses Evaluation ..... 3-57
3.4.1 Fatigue Evaluation of the Dry Shielded Canisters ..... 3-58
3.4.2 Fatigue Evaluation of the Transfer Casks ..... 3-60
3.4.3 Horizontal Storage Module Concrete and Dry Shielded Canister Steel Support Structure Thermal Fatigue, Corrosion, and Temperature Effects Evaluation ..... 3-62
3.4.4 Dry Shielded Canister Poison Plates Boron Depletion Evaluation ..... 3-64
3.4.5 Evaluation of Neutron Fluence and Gamma Radiation on Storage System Structural Materials ..... 3-65
3.4.6 Confinement Evaluation of 24P and 52B Non-Leaktight DSCs ..... 3-66
3.4.7 Thermal Performance of Horizontal Storage Modules for the Period of Extended Operation ..... 3-68
3.4.8 Evaluation of Additional Cladding Oxidation and Additional Hydride Formation Assuming Breach of Dry Shielded Canister Confinement Boundary ..... 3-69
3.4.9 Evaluation of Cladding Gross Rupture during Period of Extended Operation ..... 3-70
3.4.10 Structural Assessment of High Burnup Cladding Performance during Period of Extended Operation ..... 3-70
3.4.11 Defense-in-Depth Evaluation of Dry Shielded Canister Internal Pressures Assuming High Burnup Fuel Cladding Failure during Period of Extended ..... 3-72
3.4.12 Defense-in-Depth Structural Evaluation of Dry Shielded Canister Confinement and Retrievability Assuming High Burnup Fuel Cladding Failure during Period of Extended Operation ..... 3-73
3.4.13 Bounding Evaluation of Dry Shielded Canister with Reduced Shell Thickness Due to Chloride-Induced Stress Corrosion Cracking under Normal and Off Normal Conditions of Storage during Renewal Period ..... 3-75
3.4.14 Evaluation Findings ..... 3-76
3.5 Lead Canister Inspection ..... 3-76
3.6 Aging Management Programs ..... 3-79
3.6.1 High Burnup Fuel Aging Management Program ..... 3-79
3.6.2 DSC External Surfaces Aging Management Program ..... 3-81
3.6.3 DSC Aging Management Program for the Effects of CISCC ..... 3-83
3.6.4 Horizontal Storage Module Aging Management Program for External and Internal Surfaces ..... 3-86
3.6.5 HSM Inlets and Outlets Ventilation Aging Management Program ..... 3-87
3.6.6 Transfer Cask Aging Management Program ..... 3-92
3.6.7 Evaluation Findings ..... 3-97

4 CHANGES TO CERTIFICATE OF COMPLIANCE AND TECHNICAL SPECIFICATIONS 4-1

5 CONCLUSION ..... 5-1

6 REFERENCES ..... 6-1

# TABLES


Table Page

1.6-1 Existing Interim Staff Guidance Relevant to ISFSI Renewal ..... 1-3

2.1-1 SSCs Within and Not Within Scope of Renewal ..... 2-3
2.1-2 SSC Subcomponents Within Scope of Renewal SNF Assemblies ..... 2-7
2.1-3 SSC Subcomponents Within Scope of Renewal 24P DSC ..... 2-8
2.1-4 SSC Subcomponents Within Scope of Renewal 24P Long Cavity DSC ..... 2-8
2.1-5 SSC Subcomponents Within Scope of Renewal 24PT2 DSC ..... 2-9
2.1-6 SSC Subcomponents Within Scope of Renewal 52B DSC ..... 2-9
2.1-7 SSC Subcomponents Within Scope of Renewal 24PHB Long Cavity DSC ..... 2-10
2.1-8 SSC Subcomponents Within Scope of Renewal 61BT DSC ..... 2-10
2.1-9 SSC Subcomponents Within Scope of Renewal 32PT DSC ..... 2-11
2.1-10 SSC Subcomponents Within Scope of Renewal 24PTH DSC ..... 2-11
2.1-11 SSC Subcomponents Within Scope of Renewal 61BTH DSC ..... 2-12
2.1-12 SSC Subcomponents Within Scope of Renewal 32PTH1 DSC ..... 2-12
2.1-13 SSC Subcomponents Within Scope of Renewal 69BTH DSC ..... 2-13
2.1-14 SSC Subcomponents Within Scope of Renewal 37PTH DSC ..... 2-13
2.1-15 SSC Subcomponents Within Scope of Renewal HSM 80, HSM 102, HSM-H, HSM-202 ..... 2-14
2.1-16 SSC Subcomponents Within Scope of Renewal HSM-HS and HSM 152 ..... 2-14
2.1-17 SSC Subcomponents Within Scope of Renewal Standardized and OS1971 Type Transfer Casks ..... 2-15
2.1-18 SSC Subcomponents Within Scope of Renewal OS197L Transfer Cask ..... 2-16
2.1-19 SSC Subcomponents Within Scope of Renewal OS2001 Type Transfer Casks ..... 2-17
2.1-20 SSC Subcomponents NOT Within Scope of Renewal ..... 2-18
2.1-21 SSC Subcomponents NOT Within Scope of Renewal HSM 80, HSM 102, HSM-H, HSM-202 ..... 2-18
2.1-22 SSC Subcomponents Within Scope of Renewal HSM-HS and HSM 152 ..... 2-18
2.1-23 SSC Subcomponents NOT Within Scope of Renewal Standardized and OS1971 Type Transfer Casks – Main Assembly ..... 2-19
2.1-24 SSC Subcomponents NOT Within Scope of Renewal OS197L Transfer Cask ..... 2-19
2.1-25 Subcomponent Parts NOT Within Scope of Renewal OS2001 Type Transfer Casks – Main Assembly ..... 2-19
3.3-1 Aging Management Review Results—SFA ..... 3-3
3.3-2 Aging Management Review Results—DSC ..... 3-3
3.3-3 Aging Management Review Results—HSM-80 and HSM-102 ..... 3-7
3.3-4 Aging Management Review Results—HSM-H and HSM-202 ..... 3-8
3.3-5 Aging Management Review Results—HSM-152 ..... 3-9
3.3-6 Aging Management Review Results—HSM-HS ..... 3-10
3.3-7 Aging Management Review Results—TC ..... 3-11
3.3-8 Materials in SFA Cladding and Hardware ..... 3-15
3.3-9 Materials in HSM Subcomponents ..... 3-38
3.3-10 Transfer Cask Designs ..... 3-51

# INTRODUCTION

By letter dated November 4, 2014 (ML14309A341), as supplemented on October 16, 2015 (ML15295A354) and June 6, 2016 (ML16169A025) and September 29, 2016 (ML16279A368) (hereafter the "renewal application"), the current Certificate of Compliance (CoC) holder, AREVA Inc. (hereafter the "applicant"), applied for renewal of CoC No. 1004 for the Standardized NUHOMS® Horizontal Modular Storage System, for a period of 40 years beyond the initial certificate period. The applicant submitted the renewal application in accordance with the regulatory requirements of 10 CFR 72.240, "Conditions for spent fuel storage cask renewal." Because the renewal application was submitted more than 30 days before the Certificate's expiration date, pursuant to 10 CFR 72.240(b), this application constitutes a timely renewal. In the application, the applicant documented the technical bases for renewal of the certificate and commitments to actions for managing the potential aging effects of the systems, structures, and components (SSCs) of the dry storage system to ensure that these SSCs will maintain their intended functions during the period of operation that extends beyond the length of the term certified by the current certificate (referred to hereafter as the period of extended operation or extended storage).

The Standardized NUHOMS® System was approved under 10 CFR 72, Subpart K for storage of Spent Nuclear Fuel (SNF) in an Independent Spent Fuel Storage Installation (ISFSI) at power reactor sites to persons authorized to possess or operate nuclear power reactors under 10 CFR 50 and 10 CFR 52. The Standardized NUHOMS® System is a canister-based dry storage system (DSS) for SNF comprised of two principal components, the welded stainless steel dry storage canister (DSC) and the horizontal storage module (HSM). The system also includes a transfer cask (TC), used for DSC loading in the spent fuel building and transfer of the DSC to the storage pad.

In the renewal application, the applicant presented general information about the DSS design and a scoping analysis to determine the SSCs that are in-scope of the renewal and subject to an aging management review (AMR). The applicant further screened the in-scope SSCs to identify and describe the subcomponents that support the in-scope SSC intended function. For each in-scope SSC subcomponent, the applicant proposed either a time-limited aging analyses (TLAA) or aging management program (AMP) to assure that the SSC will maintain its intended function(s) during the period of extended operation.

The NRC staff (staff) reviewed the technical bases for safe operation of the DSS for an additional 40 years beyond the current CoC term of 20 years. This safety evaluation report (SER) summarizes the results of the staff's review for compliance with 10 CFR 72.240. In its review of the application and development of the SER, the staff followed the guidance provided in NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," dated June 2016 (NRC, 2016).

This SER is organized in five sections: Section 1 provides the staff's review of the general information of the DSS. Sections 2 and 3 document the staff's evaluation of the application and issues considered during the review of the application. Section 4 documents the changes to the CoC conditions and technical specifications being made to the initial CoC and associated amendments due to renewal. Section 5 provides the staff conclusions of the safety review.

# 1 GENERAL INFORMATION

## 1.1 CoC and CoC Holder Information

On November 4, 2014, (ML14309A341), as supplemented on October 16, 2015 (ML15295A354), June 6, 2016 (ML16169A025), and September 29, 2016 (ML16279A368), the applicant submitted an application to renew CoC No. 1004 for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, under the provisions of 10 CFR part 72, subparts K and L.

The applicant requested renewal of the initial Standardized NUHOMS® Storage System CoC and amendments 1 through 11, 13 and 14. The initial CoC (Amendment 0) was issued on January 23, 1995, for the Standardized NUHOMS® Horizontal Modular Storage System. Subsequently, thirteen amendments (1 through 11, 13 and 14) have been issued to the Standardized NUHOMS® Storage System CoC. The applicant has provided a description of the certification basis for the Standardized NUHOMS® Storage System initial issuance, and general descriptions of the changes and reasons for each amendment, including the dates of the applications and associated supplements, the date of CoC and CoC amendments issuance, and the corresponding updated final safety analysis report (UFSAR) revisions in which the changes were incorporated. A list of the amendments has been provided in Chapter 1 of the application, with details and references for each amendment in Appendices 1A through 1K.

## 1.2 Safety Review

The objective of this safety review is to determine whether there is reasonable assurance that the DSS continues to meet the requirements of 10 CFR Part 72 during the period of extended operation. The NRC staff safety review is a detailed and in-depth assessment of the technical aspects of the Standardized NUHOMS® System renewal application. Pursuant to 10 CFR 72.240(c)(2) and 72.240(c)(3), an application for renewal of a spent fuel storage cask CoC must be accompanied by a safety analysis report that includes the following: (i) time-limited aging analyses (TLAAs) that demonstrate SSCs important to safety (ITS) will continue to perform their intended function for the requested period of extended operation and (ii) a description of the aging management programs (AMPs) for management of issues associated with aging that could adversely affect ITS structures, systems and components (SSCs). The applicant stated that the renewal application is consistent with guidance provided in NUREG-1927, “Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance” (NRC, 2011a). The applicant provided both TLAAs and AMPs to assure that the SSCs within the scope of renewal will continue to perform their intended function during the period of extended operation. The staff notes that, since the submission of the original renewal application, a revision to NUREG-1927 was issued. However, the staff notes that the applicant remained informed of changes being made in the revised guidance, and revised the application as necessary throughout the review process. Therefore, this review documents the staff’s evaluation of the applicant’s scoping and screening evaluation, aging management review, and supporting AMPs and TLAAs per the guidance in NUREG-1927, Revision 1 (NRC, 2016).

## 1.3 Application Content

The renewal application provided the following information:

- General Information

- Scoping Evaluation
- Aging Management Reviews
- Aging Management Tollgates
- NUHOMS CoC 1004 Amendment Renewals
- Time-Limited Aging Analyses and Other Supplemental Evaluations
- Lead Canister Inspection
- Chloride Induced Stress Corrosion Cracking (CISCC) Information
- Aging Management Programs
- Changes to the CoC 1004 Updated Final Safety Analysis Report
- Changes to the CoC 1004 Technical Specifications

The applicant also provided UFSAR revisions for all CoC amendments, which incorporated all changes to the Standardized NUHOMS® System previously made without prior NRC approval in accordance with 10 CFR 72.48(c) and (d). The UFSAR Supplement and Changes document provided in the application documents the changes and additions for which the applicant has committed. In addition, during the staff's review of the renewal application, the applicant submitted its 2016 biennial update to the UFSAR (AREVA TN, 2016). The staff considered these additional UFSAR revisions in its review of the applicant's scoping evaluation and aging management review.

## 1.4 Interim Staff Guidance

The staff, industry, and other interested stakeholders gain experience and develop lessons learned from operating ISFSIs and DSSs, as well as each renewal review. The lessons learned address issues related to the licensing goals of maintaining safety, improving effectiveness and efficiency, reducing regulatory burden, and increasing public confidence. The staff develops Interim Staff Guidance (ISG) to clarify or to address issues not addressed in standard review plans, including NUREG-1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," Revision 1 (NRC, 2010a) and NUREG-1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel" (NRC, 2016). These ISGs are to be used by the staff, industry, and other interested stakeholders until incorporated into staff guidance documents such as regulatory guides and standard review plans. Table 1.6-1 lists the ISGs relevant to this ISFSI CoC renewal.

## 1.5 Evaluation Findings

The staff reviewed the general information provided in Chapter 1 of the renewal application. The staff performed its review following the guidance provided in NUREG-1927 and relevant ISGs, as identified in Table 1.6-1. Based on its review, the staff finds:

F1.1 The information presented in the renewal application satisfies the requirements of 72.240, "Conditions for spent fuel storage cask renewal".

F1.2 The applicant has provided a tabulation of all supporting information and docketed material incorporated by reference, in compliance with 10 CFR 72.240, "Conditions for spent fuel storage cask renewal".

| Table 1.6-1. Existing Interim Staff Guidance Relevant to CoC Renewal | |
|---|---|
| **Interim Staff Guidance Number** | **Interim Staff Guidance Title** |
| SFST-ISG-2, Rev. 1 (NRC, 2010b) | Fuel Retrievability |
| SFST-ISG-11, Rev. 3 (NRC, 2003) | Cladding Considerations for the Transportation and Storage of Spent Fuel |
| SFST-ISG-25 (NRC, 2010c) | Pressure Test and Helium Leakage Test of the Confinement Boundary for Spent Fuel Storage Canister |

# 2 SCOPING EVALUATION

10 CFR 72.240(c)(2) and (3) require a CoC renewal application to include TLAAs that demonstrate that ITS SSCs will continue to perform their intended function for the requested period of extended operation; and a description of AMPs for management of issues associated with aging that could adversely affect ITS SSCs. In addition, 10 CFR 72.122(l) requires that storage systems be designed to allow ready retrieval of spent fuel, high-level radioactive waste and reactor-related greater than class C waste (GTCC) for further processing or disposal.

A scoping evaluation is necessary to identify the SSCs subject to an AMR. More specifically, the scoping evaluation is used to identify SSCs meeting any of the following criteria:

1. SSCs that are classified as important to safety (ITS), as they are relied on to do one of the following functions:

    i. Maintain the conditions required by the regulations, specific license, or CoC to store spent fuel safely.

    ii. Prevent damage to the spent fuel during handling and storage.

    iii. Provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

2. SSCs that are classified as <u>not</u> important to safety but, according to the design bases, their failure could prevent fulfillment of a function that is important to safety.

The applicant performed a scoping evaluation consistent with the above criteria and provided the following information:

- A description of the scoping and screening methodology for the inclusion of SSCs and SSC subcomponents in the renewal scope;
- A list of the SSCs and SSC subcomponents identified to be within the scope of renewal and subject to an AMR, including their intended function, and safety classification or basis for inclusion in the renewal scope;
- A list of sources of information used; and
- Any discussion and drawings needed to clarify the process, SSC designations, or sources of information used.

The following section discusses the staff's review and review findings on the applicant's scoping study.

## 2.1 Scoping and Screening Methodology

Chapter 2 of the renewal application, Scoping Evaluation, describes the methodology for identifying SSCs within the scope of renewal and subject to an AMR. The applicant followed a scoping evaluation process in accordance with NUREG--1927, "Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance"

(NRC, 2011a). The applicant's scoping and screening methodology reviewed the design bases information as identified in the following documents:

- 72-1004 Safety Analysis Reports
    - Revision 0, December 20, 1990
    - Revision 1, September 25, 1991
    - Revision 3, November 5, 1993
    - Revision 3A, June 1995
    - Revision 4A, June 1996
    - Revision 5, September 2000
    - Revision 6, October 2001
    - Revision 7, November 28, 2003
    - Revision 8, June 2004
    - Revision 9, January 2006
    - Revision 10, February 2008
    - Revision 11, February 2010
    - Revision 12, February 2012
    - Revision 13, January 2014
    - Revision 14, September 2014
- 72-1004 SER
- Initial Certificate Effective Date: January 23, 1995
- Amendment Number 1 Effective Date: April 27, 2000
- Amendment Number 2 Effective Date: September 5, 2000
- Amendment Number 3 Effective Date: September 12, 2001
- Amendment Number 4 Effective Date: February 12, 2002
- Amendment Number 5 Effective Date: January 7, 2004
- Amendment Number 6 Effective Date: December 22, 2003

- Amendment Number 7 Effective Date: March 2, 2004
- Amendment Number 8 Effective Date: December 5, 2005
- Amendment Number 9 Effective Date: April 17, 2007
- Amendment Number 10 Effective Date: August 24, 2009
- Amendment Number 11 Effective Date: January 7, 2014
- Amendment Number 13 Effective Date: May 24, 2014
- Amendment Number 14 Effective Date: April 25, 2017

The CoC conditions and technical specifications (TS) govern the transfer of irradiated nuclear fuel from the spent fuel pool and the ISFSI, including DSS handling, loading, movement, surveillance, and maintenance of the loaded DSSs.

### 2.1.1 Scoping Process

The applicant reviewed the Standardized NUHOMS® System design-bases documents listed in Section 2.1 of this SER to identify SSCs with safety functions meeting either scoping criterions 1 or 2, as defined in Section 2. Table 2.1-1 lists the SSCs included and excluded from the scope of renewal.

**Table 2.1-1. SSCs Within and Not Within Scope of Renewal**

| Structures, Systems and Components | Criterion 1 | Criterion 2 | In-Scope |
|---|---|---|---|
| SNF Assemblies | Yes | N/A | Yes |
| Dry Shielded Canister (DSC) | Yes | N/A | Yes |
| Horizontal Storage Module (HSM) | Yes | N/A | Yes |
| Transfer Cask (TC) | Yes | N/A | Yes |
| Transfer Cask Lifting Yoke[1] | No | N/A | No |
| ISFSI Basemat[2] | No | Yes | Yes |
| ISFSI Approach Slab | No | No | No |
| Other Transfer Equipment[3] | No | No | No |
| Auxiliary Equipment[4] | No | No | No |
| Miscellaneous Equipment[5] | No | No | No |

[1] The applicant stated that the Transfer cask lifting yoke is used for handling of the transfer cask within the fuel/reactor building and is designed and procured as a "Safety Related" component as it is used by the applicant under 10 CFR Part 50. Because the lifting yokes are typically procured and used under licensees' reactor operations, they would not be expected be tied to the design bases of the storage system.

[2] The applicant stated that the Basemat is not important to safety but is included in scoping under criterion 2.

[3] The applicant stated Other Transfer Equipment includes a hydraulic ram system, a prime mover for towing, a transfer trailer, a ram support assembly, a cask support skid, auxiliary equipment mounted on the skid, and a skid positioning system.

[4] The applicant stated Auxiliary Equipment includes (but not limited to) the cask/canister annulus seal, a vacuum drying system and an automatic welding system used to facilitate canister loading, draining, drying, inerting, and sealing operations.

[5] The applicant stated Miscellaneous Equipment includes (but not limited to) ISFSI security fence and gate(s), lighting, lightning protection, communications, monitoring, and alarm systems.

The SSCs identified in Table 2.1-1 to be within the scope of renewal include:

- Spent Nuclear Fuel (SNF) Assemblies
- Dry Shielded Canister (DSC)
- Horizontal Storage Module (HSM)
- Transfer Cask (TC)
- ISFSI Basemat

The applicant stated the SNF assemblies, DSC, HSM, and the TC were found to meet scoping criterion 1 as defined in Section 2 of this SER, and hence were subject to an AMR. The applicant stated that the ISFSI basemat is not important to safety but was found to meet scoping criterion 2 and included in the scope of the review. The staff reviewed Chapter 1 of the Standardized NUHOMS System UFSAR, which includes detailed descriptions of the DSCs, HSMs, TCs, the arrangement of the structures and the type of SNF assemblies that are stored as well as the information on the DSCs, HSMs, TCs, and fuel to be stored identified in each of the NUHOMS System amendments.

Although the design bases in the UFSAR did not identify SNF assemblies as important to safety, in its renewal application, the applicant stated that the SNF assembly subcomponents have intended functions of confinement, radiation shielding, sub-criticality control, structural integrity and content temperature control. Therefore, the applicant determined that SNF assemblies were within the scope of renewal.

The staff reviewed the applicant's scoping evaluation to verify that the SNF assemblies have intended functions that must be maintained in the requested period of extended operation. Pursuant to 10 CFR 72.122(h)(1), spent nuclear fuel cladding must be protected during storage against degradation that leads to gross ruptures. Therefore, the staff notes that the cladding is within scope of renewal since it meets criterions (1)(i) and 1(ii) above. 10 CFR 72.122(l) requires that all storage systems must be designed to allow for ready retrieval of the spent fuel. 10 CFR 72.236(m) further requires that, to the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel. Per NUREG-1536, Revision 1, ready retrieval means maintaining the ability to handle individual or canned spent fuel assemblies by the use of normal means (i.e., by the use of a crane and grapple used to move undamaged assemblies at the point of cask loading). In addition to supporting retrievability, the condition of the assembly hardware and cladding provide reasonable assurance that the fuel will remain in the analyzed configuration. Therefore, the staff considers that both the assembly hardware and cladding meet criterion (1)(iii) above. Since the fuel configuration is used to define the safety analyses for confinement, radiation shielding, sub-criticality control, structural integrity and content temperature control, the staff concludes that the SNF assemblies are within the scope of renewal.

The applicant stated that the Dry Shielded Canister (DSC) and its subcomponents have intended functions of confinement, radiation shielding, sub-criticality control, structural integrity, and content temperature control and therefore, the DSCs are included within the scope of the renewal. The staff reviewed the UFSAR to identify the intended functions, and confirmed that the DSCs are designed and constructed with features and subcomponents necessary for the intended safety functions. The DSC shell assembly consists of a stainless steel welded pressure vessel that provides confinement of radioactive materials. The cylindrical shell with the integrated bottom shield plug and the top shield plug provide radiation shielding during loading and transfer operations. Criticality control is achieved by geometric separation of the fuel assemblies in the internal basket and by the use of neutron absorbing materials in the basket. Structural integrity is provided by the internal basket, shell assembly, and top and bottom cover plates. The internal basket assembly and helium backfill provide heat transfer from the stored contents to the DSC shell to provide content temperature control necessary to maintain fuel cladding temperatures during storage. Therefore the staff conclude that the DSCs must be included within the scope of the renewal.

The applicant stated that the Horizontal Storage Module (HSM) and its subcomponents have intended safety functions of radiation shielding, structural integrity and content temperature control and therefore, the HSMs is included within the scope of the renewal. The staff reviewed the UFSAR to identify the intended functions, and confirmed that the HSMs are designed and constructed with features and subcomponents necessary for the intended safety functions. Structural support of the loaded DSC is provided by a steel assembly anchored to the floor slab and walls of the HSMs. Radiation shielding is provided by thick concrete side walls and a concrete roof slab. Content temperature is controlled by heat removal, achieved by a combination of radiation, conduction, and convection. The HSMs are designed to provide shielding but still allow the intake of air through ventilation inlet openings located in the lower region of the front or side walls. The air circulates around the DSCs to provide cooling prior to exiting through outlet openings in the top regions of the HSM walls. Therefore, based on these intended safety functions, the staff conclude that the HSMs must be included within the scope of the renewal.

The applicant stated that the Transfer Cask (TC) and its subcomponents have intended safety functions of radiation shielding, structural integrity, and content temperature control and therefore, the TCs are included within the scope of the renewal. The staff reviewed the UFSAR to identify the intended functions, and confirmed that the TCs are designed and constructed with features and subcomponents necessary for the intended safety functions. Several materials including a solid neutron shielding material or water, and lead or carbon steel for gamma shielding are used depending on the transfer cask design. The TC shell assembly, bottom assembly, bolted lid provide both structural integrity and content temperature control safety functions. Therefore the staff conclude that the TCs must be included within the scope of the renewal.

The applicant stated that the Transfer Cask Lifting Yoke is used for handling of the transfer cask within the fuel/reactor building and is designed and procured as a “Safety Related” component as it is used by the general licensee under 10 CFR Part 50. Therefore, the applicant determined that the Transfer Cask Lifting Yoke is not included within the scope of the renewal. The staff reviewed the UFSAR and confirmed that the design bases of the Standardized NUHOMS System do not include the lifting yoke. The staff notes that lifting yokes typically are procured and used under licensees’ reactor operations, and thus they would not be expected be

tied to the design bases of the storage system. Therefore the staff conclude that the Transfer Cask Lifting Yoke does not need to be included within the scope of the renewal.

The applicant stated that the ISFSI basemat and approach slab are classified as not important-to-safety as they are not relied upon to provide safety functions. The HSM is installed on a load-bearing foundation, which consists of a reinforced concrete pad on a subgrade suitable to support the loads, however, there are no structural connections or means to transfer shear between the HSM base unit module and the concrete basemat. The approach slab is a reinforced concrete slab that provides access and support to the DSC transfer system. The staff reviewed the UFSAR and confirmed that the ISFSI basemat and approach slabs are not considered important to safety and are designed, constructed, maintained, and tested as commercial grade items. Although the concrete basemat is classified as not important to safety, the applicant has identified the basemat as included within the scope of the renewal by criterion 2, because settlement of the basemat may affect retrievability of the DSC. Therefore the staff conclude that the ISFSI basemat must be included within the scope of the renewal.

The applicant stated that other transfer equipment, including a hydraulic ram system to insert the DSC into or remove the DSC from the HSM, a ram support assembly, a cask support skid, auxiliary equipment mounted on the skid, a skid positioning system a transfer trailer, and a prime mover for towing the transfer trailer, are classified as not important to safety and therefore are not included within the scope of the renewal. The applicant stated that the DSC and the TC are designed to withstand potential failure of the fuel transfer equipment and would not prevent the DSC or the TC from fulfilling their intended safety functions. The staff reviewed the UFSAR and confirmed that the applicant has supplied transfer cask drop analyses that demonstrate that the performance of these items is not required to 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. Thus, failure of the other transfer equipment including the hydraulic ram system, a prime mover for towing, a transfer trailer, a ram support assembly, a cask support skid, auxiliary equipment mounted on the skid, and a skid positioning system would not prevent the DSC or the TC from fulfilling their intended functions. Therefore, the staff conclude that the other transfer equipment identified in Table 2.1-1 does not meet scoping criterion 2 and are not within the scope of renewal.

The applicant stated that Auxiliary Equipment including the cask/canister annulus seal, a vacuum drying system, and an automatic welding system used to facilitate canister loading, draining, drying, inerting, and sealing operations are classified as not important to safety and therefore are not included within the scope of the renewal. The staff reviewed the UFSAR to identify the intended functions and found that the all of the auxiliary equipment identified do not have intended safety functions. The UFSAR states that the performance of these items is not required to 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. In addition, the assessment in the UFSAR states that failure of any part of these systems may result in delay of operations, but will not result in a hazard to the public or operating personnel. Consistent with guidance in NUREG-1927, Rev. 1 (NRC, 2016), active auxiliary equipment (the cask/canister annulus seal, a vacuum drying system and an automatic welding system used to facilitate canister loading, draining, drying, inerting, and sealing operations) is not within the scope of renewal, as they are not important to safety and do not prevent fulfilment of a safety function. Therefore, the staff conclude that the auxiliary equipment identified in Table 2.1-1 does not meet scoping criterion 2 and are not within the scope of renewal.

The applicant stated that miscellaneous security equipment including security fence and gate(s), lighting, lightning protection, communications, monitoring, and alarm systems are not are not part of the CoC 1004 storage system approved in accordance with 10 CFR Part 72, Subpart L. Consistent with guidance provided in NUREG-1927, Rev. 1 (NRC, 2016), failure of the ISFSI security equipment would not prevent fulfilment of a safety function. The staff reviewed the information provided, and because the ISFSI security equipment does not meet scoping criterion 2 as it is not relied upon for the fulfillment of an important safety function for an SSC that is important to safety, the staff conclude that the miscellaneous equipment identified in Table 2.1-1 is not within the scope of renewal.

The applicant screened the in-scope SSCs to identify and describe the subcomponents that support the SSC intended function(s). The applicant identified those SSC subcomponents and associated intended functions based on the design bases documents listed in Section 2.1 of this SER.

The staff's review of the subcomponents is predicated on the understanding that subcomponents may degrade under different modes or variable rates. This consideration is important since the performance of the subcomponents could affect the performance of in scope SSC during the period of extended operation.

### 2.1.2 Structures, Systems, and Components Within Scope of Renewal

Based on the scoping process discussed in Section 2.1 of the renewal application, the applicant identified five SSCs of the Standardized NUHOMS System to be within the scope of renewal. Tables 2.1-2, through 2.1-19 describe the subcomponents that support the intended functions of the SSCs within the scope of renewal.

| **Table 2.1-2. SSC Subcomponents Within Scope of Renewal**<br>**Spent Nuclear Fuel (SNF) Assemblies** |
|---|
| Fuel Cladding and End Plugs[1, 2, 3] |
| Spacer Grid Assemblies |
| Upper End Fitting/Nozzle (and related subcomponents) |
| Lower End Fitting/Nozzle (and related subcomponents) |
| Guide Tubes |
| BWR Fuel Channel ( 52B DSC Only)[4] |
| Poison Rod Assemblies (32PT DSC Only) |
| |

[1] The applicant stated that in the AMR the cladding is considered the limiting component of the SFA hardware because it serves as a barrier to fission products, provides defense-in-depth, and its structural integrity ensures SFA retrievability.

[2] The applicant defined the material for the fuel cladding and end plugs as zirconium alloy, including Zircaloy-2, Zircaloy-4, M5®, or Zirlo™.

[3] The applicant stated that cladding of failed fuel assemblies is not able to provide any intended function and is, therefore, not considered in scope for renewal.

[4] The applicant stated that The 52B DSC accepts channeled fuel only. The channel provides the interface with the basket's spacer disk and is not credited in the structural analysis of the 52B DSC basket.

| Table 2.1-3. SSC Subcomponents Within Scope of Renewal 24P Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| DSC Shell | Guide Sleeves |
| Bottom Shield Plug[1] | Oversleeves |
| Inner Bottom Cover Plate | Spacer Disks |
| Outer Bottom Cover Plate | Support Rods |
| Grapple Ring | |
| Grapple Ring Support | |
| Siphon and Vent Block | |
| Siphon and Vent Port Cover Plates | |
| Support Ring | |
| Top Shield Plug[1] | |
| Inner Top Cover Plate | |
| Outer Top Cover Plate | |
| Weld Filler Metal | |
| [1] The applicant stated that this subcomponent is NITS but is included in the scope of the review as a Criterion 2 item | |

| Table 2.1-4. SSC Subcomponents Within Scope of Renewal 24P Long Cavity Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| DSC Shell | Guide Sleeves |
| Bottom Shield Plug Assembly with Outer Bottom Cover Plate | Oversleeves |
| Inner Bottom Cover Plate | Spacer Disks |
| Grapple Ring | Support Rods |
| Grapple Ring Support | |
| Siphon and Vent Block | |
| Siphon and Vent Port Cover Plates | |
| Support Ring | |
| Top Shield Plug Assembly | |
| Inner Top Cover Plate | |
| Outer Top Cover Plate | |
| Weld Filler Metal | |

| Table 2.1-5. SSC Subcomponents Within Scope of Renewal 24PT2 Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| DSC Shell | Guidesleeve |
| Bottom Shield Plug Assembly (24PT2L) | Oversleeves |
| Bottom Shield Plug[1] (24PT2S) | Spacer Disks |
| Inner Bottom Cover Plate | Spacer Sleeves |
| Outer Bottom Cover Plate | Support Rod |
| Grapple Ring | Shim Plate |
| Grapple Ring Support | Neutron Absorber Sheet |
| Siphon and Vent Block | |
| Siphon and Vent Port Cover Plates | |
| Support Ring | |
| Top Shield Plug Assembly (24PT2L) | |
| Top Shield Plug[1] (24PT2S) | |
| Inner Top Cover Plate | |
| Outer Top Cover Plate | |
| Weld Filler Metal | |
| [1] The applicant stated that this subcomponent is NITS but is included in the scope of the review as a Criterion 2 item | |

| Table 2.1-6. SSC Subcomponents Within Scope of Renewal 52B Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| DSC Shell | Spacer Disks |
| Bottom Shield Plug[1] | Support Rods |
| Inner Bottom Cover Plate | Spacer Sleeves |
| Outer Bottom Cover Plate | Support Bars |
| Grapple Ring | Neutron Absorbing Plates |
| Grapple Ring Support | Hardware (screws, nuts, pins) |
| Siphon and Vent Block | |
| Siphon and Vent Port Cover Plates | |
| Support Ring | |
| Top Shield Plug[1] | |
| Inner Top Cover Plate | |
| Outer Top Cover Plate | |
| Weld Filler Metal | |
| [1] The applicant stated that this subcomponent is NITS but is included in the scope of the review as a Criterion 2 item | |

| Table 2.1-7. SSC Subcomponents Within Scope of Renewal 24PHB Long Cavity Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| DSC Shell | Guide Sleeves |
| Bottom Shield Plug | Oversleeves |
| Bottom Shield Plug Forging and Plate | Spacer Disks |
| Outer Bottom Cover Plate | Support Rods |
| Grapple Ring | Top Cap Plate and Liner for Damaged Fuel Cell |
| Grapple Ring Support | Bottom Cap Plate and Liner for Damaged Fuel Cell |
| Inner Grapple Ring Support | |
| Siphon and Vent Block | |
| Siphon and Vent Port Cover Plates | |
| Support Ring | |
| Top Shield Plug Forging and Plate | |
| Top Shield Plug | |
| Top Shield Plug Cover Plate | |
| Top Shield Plug lifting post | |
| Outer Top Cover Plate | |
| Weld Filler Metal | |
| Test Port Plug | |

| Table 2.1-8. SSC Subcomponents Within Scope of Renewal 61BT Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| Canister Shell | Fuel Compartment |
| Bottom Shield Plug | Fuel Compartment Extension |
| Inner Bottom Cover Plate | Poison/Neutron Absorber plate |
| Outer Bottom Cover Plate | Hardware (Weld Stud, Washers, Nuts) |
| Grapple Ring | Basket Plate Insert |
| Grapple Ring Support | Basket rail |
| Siphon and Vent Block | Basket Holddown Ring Assembly |
| Siphon and Vent Port Cover Plates | Basket Holddown Ring Assembly for damaged fuel |
| Support Ring Segment | Top Cap Plate and Liner for Damaged Fuel Cell |
| Top Shield Plug | Bottom Cap Plate and Liner for Damaged Fuel Cell |
| Inner Top Cover Plate | Weld Filler Metal |
| Outer Top Cover Plate | |
| Test Port Plug | |
| Weld Filler Metal | |

| Table 2.1-9. SSC Subcomponents Within Scope of Renewal 32PT Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| Canister Shell | Basket Plates |
| Bottom Shield Plug | Neutron Absorbing Sheet |
| Inner Bottom Cover Plate | Basket Rails |
| Outer Bottom Cover Plate | Hardware (Studs, Washers, nuts, screws) |
| Alternate Bottom Shield Plug Outer Casing | Weld Filler Metal |
| Alternate Inner Bottom Cover Plate | |
| Alternate Shell Bottom | |
| Grapple Ring | |
| Grapple Ring Support | |
| Shear Key | |
| Siphon and Vent Block | |
| Siphon and Vent Port Cover Plates | |
| Support Ring | |
| Top Shield Plug | |
| Inner Top Cover Plate | |
| Outer Top Cover Plate | |
| Test Port Plug | |
| Weld Filler Metal | |

| Table 2.1-10. SSC Subcomponents Within Scope of Renewal 24PTH Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| Canister Shell | Poison Plate |
| Bottom Shield Plug | Basket Plate |
| Inner Bottom Cover | Weld Stud, Washer, Hex Nut |
| Outer Bottom Cover | Top and Bottom Caps |
| Grapple Ring and Support | Failed Fuel Cans |
| Top Shield Plug | Basket Rails |
| Inner Top Cover Plate | Weld Filler Metal |
| Outer Top Cover Plate | |
| Bottom Forging | |
| Siphon/vent Port Cover Plate | |
| Siphon Vent Block | |
| Support Ring | |
| Test Port Plug | |
| Weld Filler Metal | |

| Table 2.1-11. SSC Subcomponents Within Scope of Renewal 61BTH Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| Cylindrical shell | Fuel compartment tube |
| Bottom shield plug | Fuel compartment wrap |
| Inner bottom cover/forging | Poison (neutron absorbing) plate |
| Outer bottom cover/forging | Basket plate |
| Grapple ring and support | Basket Stud, washers, hex nut |
| Top shield plug | Basket plate insert |
| Inner top cover plate | R45 & R90 Basket rails, including R90 stiffener plate |
| Outer top cover plate | Basket holddown ring assembly |
| Siphon/vent port cover plate | Top grid assembly |
| Siphon vent block | Weld filler metal |
| Support ring | Top Cap for damaged fuel |
| Test port plug | Bottom cap for damaged fuel |
| Weld filler metal | Failed Fuel Can |
| Key | |

| Table 2.1-12. SSC Subcomponents Within Scope of Renewal 32PTH1 Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| Canister shell | Fuel compartment |
| Bottom shield plug | Neutron Absorber plate |
| Inner bottom cover | Basket plate |
| Outer bottom cover | Weld Stud, washer, hex nut |
| Shell Bottom | Basket plate supports (inserts) |
| Grapple ring and support | Basket rail |
| Top shield plug | Weld filler metal |
| Inner top cover plate | Top and Bottom Caps |
| Outer top cover plate | Hardware (spacers, washers)[1] |
| Siphon/vent port cover plate | |
| Siphon vent block | |
| Support ring | |
| Test port plug | |
| Lifting lug and reinforcing pad | |
| Key | |
| Weld filler metal | |

[1] The applicant stated that this subcomponent is NITS but is included in the scope of the review as a Criterion 2 item

| Table 2.1-13. SSC Subcomponents Within Scope of Renewal 69BTH Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| Cylindrical shell | Fuel compartment tube |
| Bottom shield plug | Fuel compartment wrap |
| Inner bottom cover/forging | Neutron Absorbing plates |
| Outer bottom cover/forging | Basket plate |
| Grapple ring and support | Basket Studs and hex nuts |
| Top shield plug | Basket Washers[1] |
| Inner top cover plate | Basket plate insert |
| Outer top cover plate | R45 & R90 Basket rails |
| Siphon/vent port cover plate | Basket holddown ring plate |
| Siphon vent block | Spacer pad |
| Support ring | Holddown ring alignment leg |
| Test port plug | Weld filler metal |
| Weld filler metal | Top Cap for damaged fuel |
| Key | Bottom cap for damaged fuel |
| [1] The applicant stated that this subcomponent is NITS but is included in the scope of the review as a Criterion 2 item | |

| Table 2.1-14. SSC Subcomponents Within Scope of Renewal 37PTH Dry Shielded Canister (DSC) | |
|---|---|
| Canister Assembly | Basket Assembly |
| Canister shell | Fuel compartment |
| Bottom shield plug | Neutron Absorbing plates |
| Inner bottom cover | Basket plate |
| Outer bottom cover | Basket plate supports (inserts) |
| Grapple ring and support | Basket rail |
| Top shield plug | Weld filler metal |
| Inner top cover plate | Top and Bottom Caps for damaged fuel |
| Outer top cover plate | Basket Studs, Washers, Hex Nuts[1] |
| Siphon/vent port cover plate | |
| Siphon vent block | |
| Support ring | |
| Test port plug | |
| Weld filler metal | |
| [1] The applicant stated that this subcomponent is NITS but is included in the scope of the review as a Criterion 2 item | |

| Table 2.1-15. SSC Subcomponents Within Scope of Renewal HSM 80, HSM 102, HSM-H, HSM-202 | |
|---|---|
| HSM-80 and HSM-102 | HSM-H and HSM-202 |
| HSM Base, End and Rear Walls | HSM Walls and Roof |
| HSM Floor Slab and Roof | Shield Walls (End, Rear, Corner) |
| DSC Support Structure Assembly[1] | DSC Support Structure Assembly[2] |
| Shielded Door Assembly[1] | Shielded Door Assembly[2] |
| DSC Axial Retainer[1] | DSC Axial Retainer Assembly[2] |
| Heat Shield Assemblies | Heat Shield Assemblies |
| Cask Restraint Assembly | Cask Restraint Assembly |
| End and Rear Shield Walls Attachment Hardware[1] | Inlet/Outlet Vents[2] |
| | End/Rear Shield Walls Attachment Hardware |
| | Optional HSM-202 Door |

[1] Select subcomponent parts per Table 2B-1 of the renewal application.
[2] Select subcomponent parts per Table 2B-2 of the renewal application.

| Table 2.1-16. SSC Subcomponents Within Scope of Renewal HSM-HS and HSM 152 | |
|---|---|
| HSM-HS | HSM-152 |
| HSM Walls and Roof | HSM Walls and Roof |
| Shield Walls (End, Rear, Corner) | Shield Walls (End, Rear, Corner) |
| DSC Support Structure Assembly[1] | DSC Support Structure Assembly[2] |
| Shielded Door Assembly[2] | Shielded Door Assembly[2] |
| DSC Axial Retainer Assembly[1] | DSC Axial Retainer Assembly[2] |
| Heat Shield Assemblies | Heat Shield Assemblies |
| Cask Restraint Assembly | Cask Restraint Assembly |
| Roof Attachment Assembly[1] | Inlet/Outlet Vents[2] |
| Inlet/Outlet Vents[1] | End/Rear Shield Walls Attachment Hardware[2] |
| End/Rear Shield Walls Attachment Hardware | |
| Module-to-Module Connections[1] | |

[1] Select subcomponent parts per Table 2B-3 of the renewal application.
[2] Select subcomponent parts per Table 2B-4 of the renewal application.

| Table 2.1-17. SSC Subcomponents Within Scope of Renewal Standardized and OS197[1] Type Transfer Casks | | |
|---|---|---|
| Structural Shell Assembly | Inner and Outer Assembly | Main Assembly |
| Top Flange | Inner Liner Plate | Castable Neutron Shielding Material[2] |
| Structural Shell | NSP Top and Bottom Support Ring[2] | Top Cover Plate |
| Bottom Support Ring | NSP Support Angle[2] | ¼ inch Thick Plate for Top Cover[2] |
| Bottom End Plate | Lead Gamma Shielding | Bottom Cover Plate |
| Ram Access Penetration | Neutron Shield Panel Plates[2] | Bolt with Standard Washer |
| Upper Trunnion Sleeve | Bottom Neutron Shield Plates[2] | Screw, Socket Head Cap with Washer[2] |
| Lower Trunnion Sleeve | Canister Rails[2] | Bolt |
| Helical Screw Thread Insert | | Plate |
| Lower Trunnion | | Washer for Cask Lid Bolts[2] |
| Upper Trunnion | | |
| Lower Trunnion (One Piece) | | |
| Upper Trunnion (One Piece) | | |
| Bottom End Forging | | |

[1] The Standardize Cask and OS197 Type TCs (except for the lighter-weight "L" design) are grouped as they share many common AMR results. The OS197 Type TCs include OS197, OS197H, OS197FC, OS197H FC, OS197FC-B, and OS197HFC-B TCs

[2] The applicant stated that this subcomponent is NITS but is included in the scope of the review as a Criterion 2 item

| Table 2.1-18. SSC Subcomponents Within Scope of Renewal OS197L Transfer Cask | | | |
|---|---|---|---|
| Cask Body Assembly | Light Neutron Shield Assembly | Support Skid Supplemental Shielding | Decontamination Area Cask Shielding |
| Top Flange | Top and Bottom Neutron Shield Rings | Out and Inner Shielding, Web Shielding, Side Plate Shielding, Rear Shielding, etc. | Upper Cask Shield |
| Structural Shell | Plate | Tie Down Plates[1] | Lower Cask Shield |
| Bottom Support Ring, Machined Forging | Upper and Lower Seam Plates | Bolts[1] | |
| Bottom End Plate | Inner Neutron Shield Panels | Standard Washers[1] | |
| Ram Access Penetration Ring, Machined Forging | Outer Neutron Shield Panels | | |
| Upper Trunnion | Upper and Lower Trunnion Rings | | |
| Lower Trunnion | NSP Support Angle | | |
| Castable Neutron Shielding Assembly[1] | | | |
| Screw Thread Insert | | | |
| Canister Rails[1] | | | |
| Bottom Neutron Shield Plate[1] | | | |
| Top Cover Plate | | | |
| Plates for Top Cover[1] | | | |
| Hardened Washers, Cadmium Plated[1] | | | |
| Bottom Cover Plate | | | |
| Screw, Cadmium Plated | | | |
| Screw, Socket Head Cap | | | |

[1] The applicant stated that this subcomponent is NITS but is included in the scope of the review as a Criterion 2 item

| Table 2.1-19. SSC Subcomponents Within Scope of Renewal OS200[1] Type Transfer Casks | | | |
|---|---|---|---|
| Structural Shell Assembly | Inner and Outer Assembly | Main Assembly | Internal Sleeve Design & Spacer |
| Top Flange, Machined Forging | Inner Liner Plate | Castable Neutron Shielding Material[2] | Aluminum Sleeve |
| Structural Shell | NSP Top and Bottom Support Rings[2] | Top Cover Plate | Sleeve Ring Spacer |
| Bottom Support Ring, Machined Forging | NSP Support Angle[2] | Plate for Top Cover[2] | Canister Rail |
| Bottom End Plate | Lead Gamma Shielding | Bottom Cover Plate | Button Head Cap Screw |
| Ram Access Penetration | Neutron Shield Panel[2] | Cadmium Plated Screw | Washer |
| Screw Thread Insert | Canister Rails[2] | Screw, Socket Head Cap | Shear Block |
| Air Wedge Plates[2] | Bottom Neutron Shield Plate[2] | Hardened Washer, Cadmium Plated[2] | Top and Bottom Plate |
| | Lower Trunnion | | Inner and Outer Ring |
| | Upper Trunnion | | Stiffener Plate |
| | Upper Trunnion Pad | | Lifting Bar |

[1] The OS200 Type TCs include the OS200 and OS200FC TC

[2] The applicant stated that this subcomponent is NITS but is included in the scope of the review as a Criterion 2 item

### 2.1.3 Structures, Systems, and Components NOT Within Scope of Renewal

The applicant reviewed the in-scope SSCs to identify and describe any subcomponents that do not support the SSC intended function. Tables 2.1-20 through 2.1-25 tabulate the DSC subcomponents, identified by the applicant, as not important to safety.

| Table 2.1-20. SSC Subcomponents NOT Within Scope of Renewal | |
|---|---|
| **Spent Nuclear Fuel Assemblies** | **Dry Shielded Canister** |
| Holddown Spring Retainer and Upper End Plugs | Dry Film Lubricant |
| Control Components | Swagelok Quick Disconnects |
| Fuel Rod Pellets and Other Internal Portions | Siphon Tube |
| Fuel Assembly inserts | Aluminum and Electroless Ni Coatings (Carbon Steel Spacer Discs and Shield Plugs) |
| Instrument tubes | Nickel-based Thread Lubricant; Thread Tape, or Sealant |
| Nozzle spring set | Stainless Steel Plugs/ Bolts (Non-Structural) |
| BWR water tubes | DSC Lifting Lugs |
| Hold Down Spring and Upper End Plugs | Alignment Key |

| Table 2.1-21. SSC Subcomponents NOT Within Scope of Renewal HSM 80, HSM 102, HSM-H, HSM-202 | |
|---|---|
| HSM-80 and HSM-102 | HSM-H and HSM-202 |
| DSC Support Structure Assembly[1] | DSC Support Structure Assembly[2] |
| Shielded Door Assembly[1] | Shielded Door Assembly[2] |
| DSC Axial Retainer[1] | DSC Axial Retainer Assembly[2] |
| End and Rear Shield Walls Attachment Hardware[1] | Inlet/Outlet Vents[2] |
| [1] Select subcomponent parts per Table 2B-1 of the renewal application.<br>[2] Select subcomponent parts per Table 2B-2 of the renewal application. | |

| Table 2.1-22. SSC Subcomponents Within Scope of Renewal HSM-HS and HSM 152 | |
|---|---|
| HSM-HS | HSM-152 |
| DSC Support Structure Assembly[1] | DSC Support Structure Assembly[2] |
| Shielded Door Assembly[2] | Shielded Door Assembly[2] |
| DSC Axial Retainer Assembly[1] | DSC Axial Retainer Assembly[2] |
| Roof Attachment Assembly[1] | Inlet/Outlet Vents[2] |
| Inlet/Outlet Vents[1] | End/Rear Shield Walls Attachment Hardware[2] |
| [1] Select subcomponent parts per Table 2B-3 of the renewal application.<br>[2] Select subcomponent parts per Table 2B-4 of the renewal application. | |

| Table 2.1-23. SSC Subcomponents NOT Within Scope of Renewal Standardized and OS197[1] Type Transfer Casks – Main Assembly |
|---|
| Trunnion Cover Plate |
| Stand-Off for Eyebolts |
| Clevis Eyebolt Hoist Ring |
| Tapered Pin |
| O-Ring Parker Seals |
| Shoulder Patter Eyebolt |
| Plug Plate |
| Clevis Eyebolt |
| Cover Plate |
| Swagelok, Quick Connect |
| Flat Head Socket Cap Screw |
| Flat Sheer Gasket |
| Targets, Stick-on |
| Short and Long Tapered Pin |
| [1]The OS197 Type TCs include OS197, OS197H, OS197FC, OS197H FC, OS197FC-B, and OS197HFC-B TCs |

| Table 2.1-24. SSC Subcomponents NOT Within Scope of Renewal OS197L Transfer Cask | | | |
|---|---|---|---|
| Cask Body Assembly | Light Neutron Shield Assembly | Main Assembly | Support Skid Supplemental Shielding |
| Upper Trunnion Cover Plate | Quick Connect, Elbow | Pipe | Hoist Rings |
| Plug | Relief Valve with Deflector Cap | Elbow | |
| Cover Plate | Elbow | | |
| Socket Head Cap Screw, Flat Head | | | |
| Flat Sheet Gasket | | | |
| Quick Connect, Male | | | |
| O-Ring | | | |
| Stand-Off for Eyebolts | | | |
| Hoist Ring | | | |
| Locating Pin | | | |

| Table 2.1-25. Subcomponent Parts NOT Within Scope of Renewal OS200[1] Type Transfer Casks – Main Assembly |
|---|
| Trunnion Cover Plate |
| Stand-Off for Eyebolts |
| Hoist Ring |
| Tapered Pin |
| O-Ring |
| Quick Connect, Male |
| Socket Head Cap Screw |
| Flat Sheet Gasket |
| Plug |
| Cover Plate |
| Relief Valve with Deflector Cap |
| [1] The OS200 Type TCs include the OS200 and OS200FC TC |

## 2.2 Evaluation Findings

The staff reviewed the scoping evaluation provided in Chapter 2 of the renewal application and supplemental documentation. The staff performed its review following the guidance provided in NUREG--1927, Revision 1, "Standard Review Plan for Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," and relevant ISGs, as identified in Table 1.6-1. The staff also used the information provided in NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety" (McConnell, Jr. et al., 1996), in its review as a reference for classification of components as ITS to determine the accuracy and completeness of the applicant's scoping evaluation. Based on its review, the staff finds:

F2.1 The applicant has identified all SSCs ITS and SSCs whose failure could prevent a SSC from performing its intended safety function per the requirements of 10 CFR 72.3, "Definitions", and 72.236, "Specific requirements for spent fuel storage cask approval and fabrication."

F2.2 The justification for any SSC determined not to be within the scope of the renewal is adequate and acceptable.

# 3 AGING MANAGEMENT REVIEW

## 3.1 Review Objective

The objective of the staff's evaluation of the applicant's aging management review (AMR) is to assess the proposed aging management activities (AMAs) for systems, structures and components (SSCs) determined to be within the scope of renewal. The AMR addressed aging mechanisms and effects that could adversely affect the ability of the SSCs (and associated subcomponents) to perform their intended functions during the period of renewal.

## 3.2 AMR Process

The applicant described the AMR process to be consistent with guidance provided in NUREG-1927 (NRC, 2011a). The AMR identified the aging mechanisms and effects applicable to each SSC subcomponent based on its material of construction and service environment during normal storage conditions. For each aging mechanism/effect, the applicant further identified either a TLAA or AMP to ensure the intended function of the SSC would be maintained during the period of extended operation.

The applicant stated in the renewal application that the materials of construction of the SSC subcomponents were identified through a review of pertinent design bases documents and general arrangement drawings in references listed in Section 2.1 of this SER. The applicant stated that the review was performed to identify the environmental conditions to which the SSCs are normally exposed. The applicant clarified that these conditions are based on a review of Revision 14 of the UFSAR and plant records.

The applicant stated that aging effects requiring management during the renewed certificate period are those that could cause a loss of passive SSC intended function(s). The applicant clarified that both potential aging effects, as well as aging effects that have actually occurred based upon industry operating experience, were considered. The applicant further clarified that environmental stressors that are conditions not normally experienced (such as accident conditions), or that may be caused by a design problem, were considered event-driven situations and were not characterized as sources of aging. The applicant stated that such event driven situations would be evaluated and corrective actions, if any, would be implemented at the time of the event. The applicant further stated that each subcomponent that was subjected to aging management review was evaluated to determine if the potential aging effects/mechanisms were credible considering the material, environment, and conditions of storage.

The staff reviewed the applicant's AMR process, including a description of the review process, the design bases references, and the discussion needed to clarify the AMR. Based on its review, the staff finds the applicant's AMR process acceptable because it is consistent with NUREG-1927 (NRC, 2011a) and adequate for identifying pertinent aging effects for the SSCs within the scope of renewal.

## 3.3 Aging Management Review Results: Materials, Service Environment, Aging Effects, and Aging Management Programs

Tables 3.3-1 through 3.3-7 provide the results of the applicant's AMR and the TLAAs or AMPs credited for the identified aging mechanisms and effects for SSC subcomponents within the scope of renewal, as provided in its renewal application.

| Table 3.3-1. Aging Management Review Results—Spent Fuel Assembly (SFA) | | | | | | |
|---|---|---|---|---|---|---|
| **Subcomponent** | **In Scope Classification Criterions 1 or 2** | **Materials** | **Environment** | **Aging Effect** | **AMR SER Section** | **TLAA/AMP SER Section** |
| Fuel Cladding and End Plugs | 2 | Zircaloy-2, Zircaloy-4, M5®, Zirlo™ | Inert Gas | None Identified[1] | 3.3.1 | 3.6.1 |
| Spacer Grid Assemblies | 2 | Zirconium- or Nickel-Based Alloys | Inert Gas | None Identified | 3.3.1 | N/A |
| Upper End Fitting/ Nozzle (and related subcomponents) | 2 | Stainless Steel | Inert gas | None Identified | 3.3.1 | N/A |
| Lower End Fitting/Nozzle (and related subcomponents) | 2 | Stainless Steel | Inert gas | None Identified | 3.3.1 | N/A |
| Guide Tubes | 2 | Zirconium-Based Alloys or Stainless Steel | Inert gas | None Identified | 3.3.1 | N/A |
| BWR Fuel Channel (DSC 52 only) | 2 | Zirconium-based Alloys | Inert gas | None Identified | 3.3.1 | N/A |
| Poison Rod Assemblies (32PT DSC only) | 2 | Boron Carbide | Inert Gas, Residual boron Coating | None identified | 3.3.1 | N/A |
| [1]The applicant defined creep and radial hydride reorientation as active aging mechanisms, although not expected to compromise the analyzed configuration of the spent fuel assemblies. Consistent with the guidance in Appendix D of NUREG-1927, Revision 1, the applicant proposed a surveillance and monitoring AMP to provide confirmation. | | | | | | |

| Table 3.3-2. Aging Management Review Results—Dry Shielded Canister (DSC) | | | | | | |
|---|---|---|---|---|---|---|
| **Subcomponent** | **In Scope Classification Criterion 1 or 2** | **Materials** | **Environment** | **Aging Effect/ Mechanism** | **AMR SER Section** | **TLAA/ AMP SER Section** |
| DSC Cylindrical Shell | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| DSC Cylindrical Shell | 1 | Stainless Steel | Sheltered | Loss of Material, Cracking, Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5, 3.4.13. 3.6.2, 3.6.3 |
| Outer Bottom Cover Plate | 1 | Stainless Steel | Embedded | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |

| Table 3.3-2. Aging Management Review Results—Dry Shielded Canister (DSC) | | | | | | |
|---|---|---|---|---|---|---|
| Outer Bottom Cover Plate | 1 | Stainless Steel | Sheltered | Loss of Material, Cracking, Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5, 3.6.2, 3.6.3 |
| Bottom Shield Plug | 1, 2 | Carbon Steel | Embedded/ Encased | Embrittlement | 3.3.2 | 3.4.5 |
| Grapple Ring | 1 | Stainless Steel | Sheltered | Loss of Material, Cracking, Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5, 3.4.13. 3.6.2, 3.6.3 |
| Grapple Ring Support | 1 | Stainless Steel | Sheltered | Loss of Material, Cracking, Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5, 3.4.13. 3.6.2, 3.6.3 |
| Top Shield Plug | 1, 2 | Coated carbon steel | Inert Gas | Embrittlement | 3.3.2 | 3.4.5 |
| Support Ring | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Inner Bottom Cover Plate | 1 | Stainless Steel | Embedded/ Inert gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Siphon & Vent Block | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Siphon & Vent Port Cover Plates | 1 | Stainless Steel | Embedded/ Inert gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Outer Top Cover Plate | 1 | Stainless Steel | Embedded | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Outer Top Cover Plate | 1 | Stainless Steel | Sheltered | Loss of Material, Cracking, Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5, 3.4.13, 3.6.2, 3.6.3 |
| Lead Shielding | 1 | Lead | Embedded/ Encased | None Identified | 3.3.2 | N/A |
| Bottom Shield Plug Assembly or plate | 1 | Stainless Steel | Inert Gas | Embrittlement | 3.3.2 | 3.4.5 |
| Test Port Plug | 1 | Stainless Steel | Sheltered | Loss of Material, Cracking, Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5, 3.4.13, 3.6.2, 3.6.3 |
| Bottom Shield Plug/Shell Bottom | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |

| Table 3.3-2. Aging Management Review Results—Dry Shielded Canister (DSC) | | | | | | |
|---|---|---|---|---|---|---|
| Bottom Shield Plug/Shell Bottom | 1 | Stainless Steel | Sheltered | Loss of Material, Cracking, Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5, 3.4.13, 3.6.2, 3.6.3 |
| Top Shield Plug | 1 | Stainless Steel | Embedded/ Inert Gas | Embrittlement | 3.3.2 | 3.4.5 |
| Top Shield Plug Assembly/Plates | 1 | Stainless Steel | Embedded/ Inert Gas | Embrittlement | 3.3.2 | 3.4.6 |
| Lifting Post | 1 | Stainless Steel | Embedded/ Encased | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Top Plug Post | 1 | Stainless Steel | Embedded/ Encased | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Bottom Shield Plug/ Outer Casing | 1 | Stainless Steel | Embedded | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Bottom Shield Plug/ Outer Casing | 1 | Stainless Steel | Sheltered | Loss of Material, Cracking, Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5, 3.4.13, 3.6.2, 3.6.3 |
| Shear Key | 1 | Stainless Steel | Sheltered | Loss of Material, Cracking, Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5, 3.6.2 3.6.3 |
| Lifting Lug | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Test Port Plug | 1 | Stainless Steel | Sheltered | Loss of Material, Cracking, Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5, 3.4.13, 3.6.2, 3.6.3 |
| Key | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Inner Top Cover/Shield Plug | 1 | Stainless Steel | Embedded/ Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Spacer Discs | 1 | Coated Carbon Steel or Alloy Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Guide Sleeves | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Oversleeves | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |

| Table 3.3-2. Aging Management Review Results—Dry Shielded Canister (DSC) | | | | | | |
|---|---|---|---|---|---|---|
| Support Rods | 1 | Carbon Steel or Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Shim Plates | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Neutron Absorbing Plates | 1 | Borated Stainless Steel | Embedded/ Encased | Fatigue, Embrittlement, Boron Depletion | 3.3.2 | 3.4.1, 3.4.4, 3.4.5 |
| Hardware/ Fasteners | 1 | Alloy Steel or Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Support Bar | 1 | Coated Carbon Steel or Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Top Cap Liner and Plate for damaged fuel | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Bottom Cap Liner and Plate for damaged fuel | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Basket Plates and inserts | 1 | Stainless Steel and Aluminum | Inert Gas, Embedded/ Encased | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Fuel Compartment | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Fuel Compartment extension | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Shim Plate | 1 | Aluminum | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Basket Holddown Ring Assembly | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Basket Rails | 1 | Stainless Steel and Aluminum | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Neutron Absorbing Sheet | 1 | Boron Aluminum Alloy / MMC / Boral® | Inert Gas, Embedded/ Encased | Fatigue, Embrittlement, Boron Depletion | 3.3.2 | 3.4.1, 3.4.4, 3.4.5 |
| Basket Transition Rails/Plates | 1 | Stainless Steel and Aluminum | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |

| Table 3.3-2. Aging Management Review Results—Dry Shielded Canister (DSC) | | | | | | |
|---|---|---|---|---|---|---|
| Failed Fuel Cans and associated hardware | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Fuel compartment tube | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Fuel compartment wrap | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Basket holddown ring assembly | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Top Grid Assembly | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Basket plate supports | 1 | Stainless Steel | Embedded/ Encased | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Basket Hardware (Spacers, Washers) | 2 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Alignment Block | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Damaged fuel end caps | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Basket Plates and Retention plates | 1 | Stainless Steel | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Basket Compartment Plates | 1 | Aluminum | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |
| Basket Transition Rails | 1 | Aluminum | Inert Gas | Fatigue, Embrittlement | 3.3.2 | 3.4.1, 3.4.5 |

| Table 3.3-3. Aging Management Review Results—HSM-80 and HSM-102[1, 2] | | | | | | |
|---|---|---|---|---|---|---|
| **Subcomponent** | **In Scope Classification Criterions 1 or 2** | **Materials** | **Environment** | **Aging Effect** | **AMR SER Section** | **TLAA/ AMP SER Section** |
| HSM Base Unit; Roof Slab; Rear, End Shield Walls | 1 | Reinforced Concrete | External/ Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| HSM Base Unit; Roof Slab; Rear, End Shield Walls | 1 | Reinforced Concrete | External/ Sheltered | Cracking | 3.3.3 | 3.6.4, 3.6.5 |

| Table 3.3-3. Aging Management Review Results—HSM-80 and HSM-102[1, 2] | | | | | | |
|---|---|---|---|---|---|---|
| HSM Base Unit; Roof Slab; Rear, End Shield Walls | 1 | Reinforced Concrete | External/ Sheltered | Change in Material Properties | 3.3.3 | 3.6.4 |
| Support Rail Plate | 2 | Stainless Steel, Austenitic Alloy | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Support Rail Plate | 2 | Stainless Steel, Austenitic Alloy | Sheltered | Cracking | 3.3.3 | 3.6.4, 3.6.5 |
| Assembly/ Hardware | 1, 2 | Carbon Steel, Alloy Steel | External | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Assembly/ Hardware | 1, 2 | Carbon Steel, Alloy Steel | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Shielded Door Core | 1 | Concrete | Embedded/ Encased | None | N/A | N/A |
| Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Cracking | 3.3.3 | 3.6.4, 3.6.5 |
| Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Change in Material Properties | 3.3.3 | 3.6.4, 3.6.5 |
| Embedded Assembly/ Hardware | 1, 2 | Carbon Steel, Alloy Steel | Embedded/ Encased | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |

[1]For specific subcomponent parts, see Tables 1A-12, 1B-10, 1C-14, and 1H-28 of the renewal application.
[2] HSM-202 as defined per Table 1C-14 in the renewal application.

| Table 3.3-4. Aging Management Review Results—HSM-H and HSM-202[1, 2] | | | | | | |
|---|---|---|---|---|---|---|
| **Subcomponent** | **In Scope Classification Criterions 1 or 2** | **Materials** | **Environment** | **Aging Effect** | **AMR SER Section** | **TLAA/ AMP SER Section** |
| HSM Base Walls, Roof; End, Rear and Corner Walls; Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| HSM Base Walls, Roof; End, Rear and Corner Walls; Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Cracking | 3.3.3 | 3.6.4, 3.6.5 |
| HSM Base Walls, Roof; End, Rear and Corner Walls; Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Change in Material Properties | 3.3.3 | 3.6.4, 3.6.5 |
| Support Rail Plate | 2 | Austenitic Alloy | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |

| Table 3.3-4. Aging Management Review Results—HSM-H and HSM-202[1, 2] | | | | | | |
|---|---|---|---|---|---|---|
| Support Rail Plate | 2 | Austenitic Alloy | Sheltered | Cracking | 3.3.3 | 3.6.4, 3.6.5 |
| Hardware/ Assembly | 1 | Carbon Steel, Alloy Steel | External/ Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Hardware/ Assembly | 1 | Anodized Aluminum | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Assembly/ Hardware | 1 | Stainless Steel | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Assembly/ Hardware | 1 | Aluminum | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Outlet Vent Cover | 1 | Reinforced Concrete | External/ Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Outlet Vent Cover | 1 | Reinforced Concrete | External/ Sheltered | Cracking | 3.3.3 | 3.6.4, 3.6.5 |
| Outlet Vent Cover | 1 | Reinforced Concrete | External/ Sheltered | Change in Material Properties | 3.3.3 | 3.6.4, 3.6.5 |

[1] For specific subcomponent parts, see Tables 1F-14, 1G-18, 1H-30, 1J-17 in the renewal application.
[2] Per Table 1G-19 in the renewal application, the intended functions and AMR results for HSM Model 202 are identical to those listed for HSM-H in Tables 1F-14 and 1G-18 .

| Table 3.3-5. Aging Management Review Results—HSM-152[1] | | | | | | |
|---|---|---|---|---|---|---|
| **Subcomponent** | **In Scope Classification Criterions 1 or 2** | **Materials** | **Environment** | **Aging Effect** | **AMR SER Section** | **TLAA/ AMP SER Section** |
| HSM Base Walls and Roof; End, Rear and Corner Walls; Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| HSM Base Walls and Roof; End, Rear and Corner Walls; Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Cracking | 3.3.3 | 3.6.4, 3.6.5 |
| HSM Base Walls and Roof; End, Rear and Corner Walls; Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Change in Material Properties | 3.3.3 | 3.6.4, 3.6.5 |
| Assembly/ Hardware | 1, 2 | Carbon Steel, Alloy Steel | External | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Support Rail | 1 | Stainless Steel | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Support Rail Plate | 2 | Austenitic Alloy | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Support Rail Plate | 2 | Austenitic Alloy | Sheltered | Cracking | 3.3.3 | 3.6.4, 3.6.5 |

| Table 3.3-5. Aging Management Review Results—HSM-152[1] | | | | | | |
|---|---|---|---|---|---|---|
| Embedded Hardware/ Assembly | 1, 2 | Carbon Steel, Alloy Steel | Embedded/ Encased | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Assembly/ Hardware | 1 | Carbon Steel, Alloy Steel | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Assembly/Hardware (Heat Shield, Zee Bracket) | 1 | Stainless Steel | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| [1] For specific subcomponent parts, see Tables 1F-15 and 1H-29 of the renewal application. | | | | | | |

| Table 3.3-6. Aging Management Review Results—HSM-HS[1] | | | | | | |
|---|---|---|---|---|---|---|
| **Subcomponent** | **In Scope Classification Criterions 1 or 2** | **Materials** | **Environment** | **Aging Effect** | **AMR SER Section** | **TLAA/ AMP SER Section** |
| HSM Base, Walls and Roof; End, Rear and Corner, Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| HSM Base, Walls and Roof; End, Rear and Corner, Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Cracking | 3.3.3 | 3.6.4, 3.6.5 |
| HSM Base, Walls and Roof; End, Rear and Corner, Shielded Door | 1 | Reinforced Concrete | External/ Sheltered | Change in Material Properties | 3.3.3 | 3.6.4, 3.6.5 |
| Rail Extension Plate | 1 | Low Alloy Steel | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Support Rail Plate | 2 | Austenitic Alloy | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Embedded Hardware/Assembly | 1 | Carbon Steel | Embedded/ Encased | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Hardware/Assembly | 1 | Carbon Steel | External/ Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Hardware/ Assembly | 1, 2 | Carbon Steel | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Hardware/ Assembly (Dose Reduction) | 2 | Stainless Steel | Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Hardware/ Assembly | 1 | Carbon Steel | External | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |

| Table 3.3-6. Aging Management Review Results—HSM-HS[1] | | | | | | |
|---|---|---|---|---|---|---|
| Outlet Vent Cover | 1 | Reinforced Concrete | External/ Sheltered | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| Outlet Vent Cover | 1 | Reinforced Concrete | External/ Sheltered | Cracking | 3.3.3 | 3.6.4, 3.6.5 |
| Outlet Vent Cover | 1 | Reinforced Concrete | External/ Sheltered | Change in Material Properties | 3.3.3 | 3.6.4, 3.6.5 |
| Hardware/Assembly (Heat Shield) | 1 | Stainless Steel | External | Loss of Material | 3.3.3 | 3.6.4, 3.6.5 |
| [1] For specific subcomponent parts, see Tables 1H-22, 1I-12, 1J-17 of the renewal application. | | | | | | |

| Table 3.3-7 Aging Management Review Results—Transfer Cask (TC)[1] | | | | | | |
|---|---|---|---|---|---|---|
| **Subcomponent** | **In Scope Classification Criterions 1 or 2** | **Materials** | **Environment[2]** | **Aging Effect** | **AMR SER Section** | **TLAA/ AMP SER Section** |
| Top Flange | 1 | Stainless Steel | External/ Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Structural Shell | 1 | Steel | Embedded/ Encased | None Identified | 3.3.5 | N/A |
| Structural Shell | 1 | Stainless Steel | Encased/ Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Bottom Support Ring, End Plates, and End Forgings | 1, 2 | Stainless Steel | External/ Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Ram Access Penetration | 1 | Stainless Steel | External/ Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Upper and Lower Trunnion Sleeves | 1 | Steel | Encased/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Upper and Lower Trunnions | 1 | Stainless Steel | External/ Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Screw Thread Insert | 1 | Stainless Steel | Embedded/ Encased | Loss of Material | 3.3.5 | 3.6.6 |
| Inner Liner Plate | 1 | Stainless Steel | Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |

| Table 3.3-7 Aging Management Review Results—Transfer Cask (TC)[1] | | | | | | |
|---|---|---|---|---|---|---|
| **Subcomponent** | **In Scope Classification Criterions 1 or 2** | **Materials** | **Environment[2]** | **Aging Effect** | **AMR SER Section** | **TLAA/ AMP SER Section** |
| Top and Bottom Neutron Shield Rings | 1, 2 | Stainless Steel | External/ Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Neutron Shield Support Angles | 2 | Stainless Steel | Embedded/ Encased | None Identified | 3.3.5 | N/A |
| Neutron Shield Support Angles | 1, 2 | Stainless Steel | Encased/Water | Loss of Material | 3.3.5 | 3.6.6 |
| Lead Gamma Shielding | 1 | Lead | Embedded/ Encased | None Identified | 3.3.5 | N/A |
| Neutron Shield Plates | 1, 2 | Stainless Steel | Embedded/ Encased/ External/ Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Neutron Shield Plates | 1 | Stainless Steel | Sheltered/ Water | None Identified | 3.3.5 | N/A |
| Canister Rails | 1, 2 | Stainless Steel | Sheltered/ Water | Loss of Material (Wear) | 3.3.5 | 3.6.6 |
| Neutron Shielding Material | 2 | NS-3 | Embedded/ Encased | None Identified | 3.3.5 | N/A |
| Top Cover Plate | 1 | Steel | External/ Sheltered | Loss of Material | 3.3.5 | 3.6.6 |
| Top Cover Plate | 1, 2 | Stainless Steel | External/ Sheltered | Loss of Material | 3.3.5 | 3.6.6 |
| Bottom Cover Plate | 1 | Stainless Steel | External/ Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Fastener Components | 1, 2 | Steel (including cadmium plated) | External/ Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Fastener Components | 1, 2 | Stainless Steel | Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Plate | 1 | Stainless Steel | Sheltered | Loss of Material | 3.3.5 | 3.6.6 |
| Washers for Cask Lid Bolts | 2 | Stainless Steel or Steel (Cadmium Plated) | External/ Sheltered | Loss of Material | 3.3.5 | 3.6.6 |

| Table 3.3-7 Aging Management Review Results—Transfer Cask (TC)[1] | | | | | | |
|---|---|---|---|---|---|---|
| **Subcomponent** | **In Scope Classification Criterions 1 or 2** | **Materials** | **Environment[2]** | **Aging Effect** | **AMR SER Section** | **TLAA/ AMP SER Section** |
| Support Skid Supplemental Shielding, Plates (OS197L TC) | 1, 2 | Steel | External/ Sheltered | Loss of Material/ Coating | 3.3.5 | 3.6.6 |
| Internal sleeve components (OS200 TC) | 1 | Aluminum | Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Internal sleeve components (OS200 TC) | 1 | Aluminum | Embedded/ Encased | None | 3.3.5 | N/A |
| Spacer plates, rings, lifting bar (OS200 TC) | 1 | Stainless Steel | Sheltered/ Water | Loss of Material | 3.3.5 | 3.6.6 |
| Spacer plates, rings, lifting bar (OS200 TC) | 1 | Stainless Steel | Embedded/ Encased | None | 3.3.5 | N/A |

[1] The transfer cask aging management review results include those from the standardized cask, OS197 type TCs (including OS197H, OS197FC, OS197H FC, OS197FC-B, and OS197HFC-B TCs), OS197L type TC, and the OS200 type TCs (including OS200 FC TC).

[2] The applicant identified three environments to which the casks are exposed. The "sheltered" environment is indoors or otherwise protected from direct exposure to sun, wind, or precipitation. The "embedded or encased" environment is one in which a material surface is sealed inside another. The "water" environment reflects the intermittent exposure of components to either spent fuel pool water during fuel loading or to neutron shield water during loading campaigns (for those casks that use a water neutron shield).

### 3.3.1 Spent Fuel Assembly (SFA)

The applicant stated that the Standardized NUHOMS® Horizontal Modular Storage System is designed to store pressurized water reactor (PWR) and boiling water reactor (BWR) spent fuel assembly (SFA) types as authorized contents per the CoC Technical Specifications.

The applicant further stated that the intended functions of the SFAs include criticality control, pressure boundary, structural integrity, and heat transfer. The applicant clarified that the geometry of the SFA is a factor in the proper conduction and convection of heat to the DSC surface and in the criticality model. The fuel cladding provides a confinement barrier, and its structural integrity is necessary to maintain a favorable geometry and for retrieval. After fuel loading and DSC drying, the SFAs are not moderated, ensuring subcriticality during subsequent operations and configurations. The SFA principal function during dry storage is to maintain proper geometry and position of radioactive material through confinement. The applicant stated that the scoping evaluation for the SFA included the fuel cladding and end plugs, guide tubes, grid assemblies, upper nozzle, and bottom nozzle, which perform the aforementioned intended functions.

The applicant identified the following SFA subcomponents to be within the scope of renewal:

- Fuel cladding
- Spacer grid assemblies
- Upper end fitting/nozzle (and related subcomponents)
- Lowed end fitting/nozzle (and related subcomponents)
- Guide tubes
- BWR fuel channel (for DSC 52B design only)
- Poisson rod assemblies (for DSC 32PT designs only)

The applicant excluded the following SFA subcomponents from the scope of renewal since they were identified to not support an intended safety function of the DSC:

- SFA inserts (e.g., burnable poison rod assemblies or thimble plug devices)
- Fuel pellets and other fuel rod internals (fuel rod spring)
- Instrument tube
- Nozzle spring set
- BWR water tubes
- Hold Down Spring and Upper End Plugs
- Control Components

The staff agrees that the subcomponents of the SFAs assure that the spent fuel remains in the analyzed configuration consistent with the confinement, structural, criticality, and thermal safety analyses in the application, as well as assuring retrievability of the SFAs. Based on its review, the staff concludes that the description of the SFAs is correct and acceptable.

#### 3.3.1.1 **Materials and Environments**

Materials

The applicant stated that the materials of construction of the SFA hardware consist of zirconium-based alloys, stainless steel, and nickel-based alloys. The applicant clarified that the AMR for the SFAs focuses primarily on the fuel rod cladding. It is considered the limiting component of the fuel assembly hardware because it serves as a barrier to fission products, provides defense-in-depth, and maintenance of its structural integrity ensures its retrievability from the DSC.

The applicant provided a general description of the SFA subcomponents, which identified the respective materials of construction and summarized below.

**Table 3.3-8. Materials in SFA Cladding and Hardware**

| HSM Subcomponents | Material |
|---|---|
| Fuel cladding and end plugs | Zirconium based alloys (Zircaloy-2, Zircaloy-4, M5®, or Zirlo™) |
| Spacer grid assemblies | Zirconium- or nickel-based alloys |
| Upper end fitting/nozzle (and related subcomponents) | Stainless steel |
| Lower end fitting/nozzle (and related subcomponents) | Stainless steel |
| Guide tubes | Zirconium-based alloys or stainless steel |
| BWR fuel channel (DSC 52B design only) | Zirconium-based alloys |
| Poison rod assemblies (DSC 32PT design only) | Boron carbide |

The applicant provided a summary of the SFAs in the renewal application. The summary of the SFA characteristics is based on DSC, design-bases heat load, average assembly initial enrichment, burnup, and cladding material. The table also clarified if damaged fuel may be stored in the corresponding DSC. The staff verified the information presented in the renewal application with the approved design bases as documented in the UFSAR.

The staff verified that the applicant's stated SFA characteristics and materials of construction for the standard and long cavity DSC 24P and the 24PT2 corresponded with those described in the UFSAR. The UFSAR identifies the fuel cladding as Zircaloy-clad fuel with no known or suspected gross cladding breaches. The design bases used B&W 15x15 fuel as the enveloping fuel design for a wide range of PWR fuel types as it is the most reactive and has the most limiting physical characteristics. Each of the DSC 24P and the 24PT2 designs may store up to 24 PWR fuel assemblies.

The staff verified that the applicant's stated SFA characteristics and materials of construction for the DSC 52B design corresponded with the descriptions in the UFSAR. The UFSAR identifies the fuel cladding as Zircaloy-clad fuel with no known or suspected gross cladding breaches. The design bases used GE 7x7 fuel as the enveloping fuel design for a wide range of BWR fuel types for the DSC 52B. The design can store up to 52 BWR fuel assemblies, provided these meet the requirements of the Technical Specifications of CoC 1004. The applicant identified the BWR fuel channel as within scope of renewal since its failure may affect the safety function of criticality control. The UFSAR identifies the fuel channel material as Zircaloy-4.

The staff verified that the applicant's stated SFA characteristics and materials of construction for the DSC 24PHBS and 24PHBL designs corresponded with the descriptions in the UFSAR. The UFSAR states that the 24PHB DSC is qualified for storage of up to 24 intact (or up to 4 damaged and balance intact) PWR fuel assemblies. The UFSAR identifies the materials for the PWR spent fuel assemblies as Zircaloy-4 (cladding material, end caps, guide tubes, guide tube plugs, grid supports), Inconel-718 (top nozzle hold down spring, upper end spring, end grids, spacer grids), 304-type stainless steel (top nozzle and bottom nozzle fittings, lower end fitting).

The staff verified that the applicant's stated SFA characteristics and materials of construction for the DSC 32PTS-100, 32PTS-125, 32PTL-100 and 32PTL-125 designs corresponded with those defined in the UFSAR. The UFSAR states that the DSC 32PT system is designed to store intact standard PWR fuel assemblies with or without control components. The UFSAR further states that the fuel cladding material is Zircaloy and cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact PWR Fuel." The UFSAR identifies the materials for the PWR spent fuel assemblies as Zircaloy (cladding material, guide tubes, instrument tubes, grid supports), Inconel-718 (top nozzle hold down spring, spacers, bottom spacer), stainless steel (top nozzle and bottom nozzle fittings, plenum springs, top nozzle spring retainer). The UFSAR also states that, if required, poison rod assemblies (PRAs) are also used for criticality control. The UFSAR clarifies that these features are only necessary during the loading and unloading operations that occur in the loading pool (underwater). However, the PRAs are left in place following the completion of the DSC draining and drying operations. The UFSAR further clarifies that, during storage, with the DSC cavity dry and sealed from the environment, criticality control measures within the installation are not necessary because of the low reactivity of the fuel in the dry DSC 32PT and the design bases requirement that no water can enter the DSC cavity during storage. The applicant identified the PRAs to be within the scope of renewal since their failure may impact the safety function of criticality control.

The staff verified that the applicant's stated SFA characteristics and materials of construction for the DSC 61BT design corresponded with those defined in the UFSAR. The UFSAR states that the DSC 61BT is designed to store 61 intact, or up to 16 damaged and remainder intact, for a total of 61, standard Boiling Water Reactor (BWR) fuel assemblies with or without fuel channels. The UFSAR identifies the cladding material for BWR fuel to be stored in the DSC 61BT design as Zircaloy. The UFSAR further identifies that cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact BWR Fuel." The UFSAR defines the cladding material for damaged BWR fuel to be stored in the DSC 61BT design as Zircaloy. The UFSAR states that damaged fuel shall be stored in compartments of the "Type C" DSC 61BT. The UFSAR identifies the materials for the BWR spent fuel assemblies as Zircaloy-2 (cladding material, end caps, channel sleeves, spacers), Inconel-750 (fuel zone spacer springs, top end expansion springs, bottom end finger springs, channel spring and bolt), 304-type stainless steel (plenum springs, tie plates, washers and nuts; channel spacer, rivet and guard). The UFSAR identifies the materials of construction for the DSC 61BT canister for BWR failed fuel cans as stainless steel.

The staff verified that the applicant's stated SFA characteristics and materials of construction for the DSC 24PTH-S, 24PTH-L, 24PTH-S-LC designs corresponded with the descriptions in the UFSAR. The UFSAR states that the 24PTH DSC is designed to accommodate up to 24 intact or up to 12 damaged (with up to 8 failed fuel cans loaded with failed fuel) with the remainder intact, PWR fuel assemblies with or without control components. The UFSAR identifies the materials for the PWR spent fuel assemblies as Zircaloy-4 (cladding material, end caps, guide tubes, grid supports), Inconel-718 (top nozzle hold down spring, upper end spring, end grids, spacer grids), 304-type stainless steel (top nozzle and bottom nozzle fittings, stainless steel rods and lower end fitting). The applicant stated that the 24PTHF is an alternate version of the 24PTHS or 24PTHL designed to accommodate failed fuel. The UFSAR describes the failed fuel can as constructed of sheet metal and is provided with a welded bottom closure and a removable top closure which allows lifting of the failed fuel can with the enclosed failed fuel. The UFSAR identifies the materials of construction for the 24PTHF failed fuel can as stainless steel.

The staff verified that the applicant's stated SFA characteristics and materials of construction for the DSC 61BTH Type 1 and Type 2 designs corresponded with the descriptions in the UFSAR. The UFSAR states that the 61BTH system is designed to store up to 61 intact (including reconstituted) or up to 16 damaged with up to 4 failed fuel cans loaded with failed fuel with the remainder intact BWR fuel assemblies with or without fuel channels. The UFSAR identifies the materials for the BWR spent fuel assemblies as Zircaloy (end plugs), Zircaloy-2 (cladding material), Zircaloy-4 (fuel channel sleeve, fuel channel, grid spacers), Inconel-750 (top nozzle expansion springs, spacer springs, channel spring and bolt), stainless steel (upper and lower tie plate, hardware, channel hardware, plenum springs). The 61BTH Type 2 DSC is designed to accommodate failed fuel. The UFSAR identifies the materials of construction for the 61BTHF failed fuel can as stainless steel.

The staff verified that the applicant's stated SFA characteristics and materials of construction for the DSC 32PTH1-S, 32PTH-M, and 32PTH designs corresponded with the descriptions in the UFSAR. The UFSAR states that the 32PTH1 system is designed to store up to 32 intact (including reconstituted) B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14, and WE 14x14 class PWR fuel assemblies. The 32PTH1 DSCs can also accommodate up to a maximum of 16 damaged fuel assemblies in lieu of an equal number of intact assemblies in the cells located at the center of the 32PTH1 basket. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of 3.0 years. There are three alternate design configurations for the 32PTH1 DSC depending on the canister length: a short DSC designated as 32PTH1-S, a medium DSC designated as 32PTH1-M, and a long DSC designated as 32PTH1-L DSC. Each of the DSC configurations is designed to store intact (including reconstituted) and/or damaged PWR fuel assemblies as specified in the UFSAR. The UFSAR identifies the materials for the PWR spent fuel assemblies as Zircaloy-4 (cladding material, end caps, guide tubes, grid supports), Inconel-718 (top nozzle hold down spring, upper end spring, end grids, spacer grids), and 304-type stainless steel (top nozzle and bottom nozzle fittings, stainless steel rods, lower end fitting, stainless steel rods).

The staff verified that the applicant's stated SFA characteristics and materials of construction for the DSC 69BTH designs corresponded with the descriptions in the UFSAR. The UFSAR states that the 69BTH system is designed to store up to 69 intact (including reconstituted) or up to 24 damaged and balance intact BWR fuel assemblies. The fuel to be stored is limited to a maximum initial lattice average initial enrichment of 5.0 wt.%, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of 3.0 years. The UFSAR further states

that the design bases for the 69BTH design allows for reconstituted fuel assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or 69 lower enrichment $UO_2$ rods instead of Zircaloy clad enriched $UO_2$ rods. The design can also accommodate damaged fuel assemblies. The UFSAR identifies the materials for the BWR spent fuel assemblies as Zircaloy (end plugs), Zircaloy-2 (cladding material), Zircaloy-4 (fuel channel sleeve, fuel channel, grid spacers), Inconel-750 (top nozzle expansion springs, spacer springs, channel spring and bolt), Inconel (finger springs), and stainless steel (upper and lower tie plate, hardware, channel hardware, plenum springs).

The staff verified that the applicant's stated SFA characteristics and materials of construction of DSC 37PTH-S and 37PTH-M designs corresponded with the descriptions in the UFSAR. The UFSAR states that the 37PTH system is designed to store up to 37 intact (including reconstituted) PWR fuel assemblies. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt.% U-235, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of 3.0 years. The design bases for the 37PTH design allows for reconstituted fuel assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or unlimited lower enrichment $UO_2$ rods instead of Zircaloy clad enriched $UO_2$ rods. The design can also accommodate damaged fuel assemblies in lieu of an equal number of intact assemblies. The UFSAR identifies the materials for the PWR spent fuel assemblies as Zircaloy-4 (cladding material, end caps, guide tubes, grid supports), Inconel-718 (top nozzle hold down spring, upper end spring, end grids, spacer grids), and 304-type stainless steel (top nozzle and bottom nozzle fittings, stainless steel rods, lower end fitting, stainless steel rods).

In sum, the staff verified that the applicant adequately defined the materials of construction of all SFA subcomponents identified to be within the scope of renewal with the descriptions in the approved design bases.

Environments

The applicant defined two environments that the SFAs experience normally and continuously: external and internal.

The applicant defined the internal environment of the SFA cladding as pressurized helium added during the manufacturing process and fission products produced during reactor operation.

The applicant identified the external environment of the SFAs to refer to the internal DSC atmosphere. More specifically, the environment was defined to be of an inert gas (helium) with trace amounts of water vapor and air. Additionally, the applicant stated that boric acid residue may coat the PWR SFA surfaces, since they were exposed to a borated water environment in the spent fuel pool prior to storage. Any boric acid residue remaining on the SFAs will have no deleterious effects due to the absence of water and due to the materials of construction for the SFAs. The applicant clarified that the helium gas temperature and pressure at the beginning of storage are governed by the design-bases maximum fuel cladding temperature and the helium backfill pressure. Both the temperature and pressure are expected to decrease as a result of the decrease in decay heat during the period of extended operation.

The renewal application lists the maximum temperatures at the beginning of storage for SFAs loaded in the various DSCs. The applicant further clarified that the allowable cladding temperature limits for SFAs loaded in the various DSCs are based on the criteria in previous

DOE reports (Johnson, et al., 1983; Levy, et al., 1987) for the earlier DSCs (e.g. 24P, 52B), or either ISG-11, Rev. 2 (NRC, 2002) or Rev. 3 (NRC, 2003) for the 32PT, 61BT, 24PTH, 61BTH, 32PTH1, 37PTH and 69BTH DSCs.

The staff verified the maximum temperature limits for the all DSCs correspond with those defined in the UFSAR. The staff agrees that the temperature and pressure of the SFAs are expected to decrease over time since the heat generated by the radioactive decay of spent fuel decreases with time. The staff concludes that the applicant adequately identified the service environment of the spent fuel assemblies.

#### 3.3.1.2 Aging Effects/Mechanisms for the Spent Nuclear Fuel Assemblies During the Period of Extended Storage

The applicant stated that cladding creep is a potentially active degradation mechanism during the period of extended operation. The relatively high initial temperatures, differential pressures, and corresponding hoop stress on the cladding could potentially result in permanent creep deformation of the cladding over time. The applicant cited testing conducted to assess creep performance of cladding under long-term storage. The cited data supports the conclusion that the spent fuel cladding has significant creep capacity even after 15 years of dry storage. The creep tests of fuel rods indicated that the creep deformation was uniform around the circumference of the cladding with no signs of localized bulging, which can be a precursor to rupture.

The staff reviewed the technical bases cited in the renewal application for identifying potential fuel and cladding degradation mechanisms, and have determined that the applicant's conclusions regarding creep are consistent with the guidance in ISG-11, Rev. 3 (NRC, 2003) and NUREG-1536, Revision 1 (NRC, 2010a). More specifically, the data and analyses support the conclusions that: (1) deformation caused by creep will proceed slowly over time and will decrease the rod pressure; (2) the decreasing cladding temperature also decreases the hoop stress, which also slows the creep rate so that during later stages of dry storage, further creep deformation will become exceedingly small; and (3) in the unlikely event that a breach of the cladding due to creep occurs, it is believed that this will not result in gross rupture.

The applicant also concluded that hydride embrittlement of the zirconium-based cladding is a potential degradation mechanism. The applicant further cited the DOE-sponsored program that evaluated the thermal performance of a CASTOR dry storage system, also known as the Dry Cask Storage Characterization Project. The fuel assemblies placed in this DSS were exposed to six thermal cycles; the two hottest cycles reached fuel cladding temperatures of 415C and 398C. The applicant stated that the fuel loaded in the CoC 1004 is similar to the Westinghouse 15x15 assembly examined, which had an assembly-averaged burnup of 35.7 GWd/MTU. The applicant stated that detailed examination of the spent fuel assemblies in the DSS showed no deleterious effects, such as fission gas release, cladding creep, cladding hydride reorientation, or cladding property degradation [NUREG CR-6745 (Bare and Torgerson, 2001) and NUREG CR-6831 (Einziger et al., 2003)]. The applicant clarified that the Dry Cask Storage Characterization Project only evaluated low burnup fuel (≤ 45 GWd/MTU), and the results may not be fully representative of the aging characteristics of high burnup fuel (> 45 GWd/MTU).

The applicant also cited ISG-11, Rev. 3, which defines the acceptance criteria needed to provide reasonable assurance that commercial spent fuel is maintained in the configuration that is analyzed in the UFSAR. The applicant stated that the basis for storage of high burnup fuel for the initial 20-year storage term is contained in ISG-11, Rev. 3. However, beyond 20 years, the

applicant cited new research indicating that hydrides could reorient at a significantly lower stress than previously believed (Kamimura, 2010; Daum et al., 2006), and that radial hydrides could raise the cladding's ductile-to-brittle transition enough to affect the ability of the cladding to withstand stress without undergoing brittle failure (Billone et al., 2013). The applicant stated that the DOE-sponsored High Burnup Dry Storage Cask Research and Development Project (HDRP) is intended to provide an experimental confirmatory basis for storage of the high burnup fuel beyond the 20 year similar to the DOE-sponsored Dry Cask Storage Characterization Project. The HDRP will characterize four (4) fuel cladding types, namely Zircaloy-4, low tin Zircaloy-4, ZIRLO™, and M5®. These cladding types are representative of the various alloying elements as well as the final metallurgical treatment (e.g., cold-worked, stress relieved, partially recrystallized, fully recrystallized) in the allowable contents for this CoC. The candidate assemblies' average assembly burnup will range between 50 GWD/MTU and 67 GWD/MTU. The applicant further cited additional details of the HDRP.

The applicant concluded that low to moderate burnup fuel is not impacted by hydrogen embrittlement and radial hydride formation, per the guidance of ISG-11, Rev. 3. The applicant based this conclusion on the results of the Dry Cask Storage Characterization Project supporting the conclusion that low burnup SFAs will not degrade under extended storage. The applicant further concluded that hydrogen embrittlement and radial hydride formation are possible degradation mechanisms for high burnup fuel cladding.

The staff verified the maximum temperature limits for all DSCs with those defined in the UFSAR. The staff confirmed the adequacy of these limits based on the technical basis discussed in ISG 11, Rev. 3.

The staff further reviewed the cited references indicating potential reduced ductility of high burnup fuel cladding due to hydride reorientation. The staff's expectation is that hydride-induced embrittlement could potentially compromise the ability to maintain the analyzed fuel configuration solely during pinch-type loads. These loads would only be possible during fuel retrieval operations, if the design bases of the DSS or ISFSI rely on retrievabilty of the high burnup fuel on a single-assembly basis. These pinch-type loads are not expected to be present during normal, off-normal, and accident conditions of storage. More specifically, the tensile stress field associated with potential inertial rod bending during storage is expected to be parallel to both radial and circumferential hydrides and not expected to compromise the structural integrity of the cladding. The NRC is sponsoring confirmatory research to this effect at Oak Ridge National Laboratory, and the results will be publicly available soon [see NUREG CR-7198 (Wang et al., 2015) for details on the experimental protocol].

The applicant also identified the potential for boron-10 (B-10) depletion in the poison rod assemblies, and provided a time-limited analysis to demonstrate adequate performance (see Section 3.5 of this SER).

The staff reviewed the identified aging mechanisms and effects for the SFAs. The staff determined that the aging management review of the spent fuel assemblies to be comprehensive and acceptable based on the design bases described in the UFSAR. Based on its review, the staff finds the applicant's identification of aging mechanisms and effects for the SFAs consistent with ISG-11, Rev. 3 guidance and therefore acceptable.

#### 3.3.1.3 Evaluation of the Proposed Aging Management Activities

The applicant stated that there are no aging effects requiring management for SFAs with assembly-average burnup ≤ 45 GWd/MTU. Therefore, no aging management program or activities are credited during the renewed CoC period for low burnup SFA subcomponents. In NUREG-1927, Revision 1, however, the staff has stated that additional confirmatory data or a commitment to obtain data on high burnup fuel, combined with appropriate steps in a learning aging management plan (AMP), will provide further information that will assure the storage and retrievability of high burnup fuel for extended durations beyond 20 years. The applicant stated that the High Burnup Fuel AMP provides confirmatory data that SFAs containing high burnup fuel will maintain their configuration during the period of extended operation. Staff review of the High Burnup Fuel AMP is included in SER section 3.6.1.

The applicant provided supporting calculations and analyses in the renewal application as a result of the AMR of the SFAs, namely: "Structural Assessment of High Burnup Cladding Performance during Period of Extended Operation"; "Evaluation of Additional Cladding Oxidation and Additional Hydride Formation Assuming Breach of Dry Shielded Canister Confinement Boundary"; and "Evaluation of Cladding Gross Rupture during Period of Extended Operation." The staff determined that these calculations or analyses do not have all attributes of a TLAA, since these were not contained or incorporated by reference in the approved design bases. Therefore, the staff reviewed these calculations and analyses in support of the proposed High Burnup Fuel AMP described in the renewal application. The staff reviewed the aging management review of the spent fuel assemblies and references therein, including design bases and applicable research results, and concluded that an AMP is an acceptable means for ensuring that the identified aging effects will not lead to a loss of intended function.

### 3.3.2 Dry Shielded Canisters (DSC)

The applicant stated that each DSC includes a shell assembly and an internal basket assembly. The DSC shell assembly is a stainless steel welded pressure vessel that consists of a cylindrical shell and the top and bottom end assemblies, which form the pressure retaining confinement boundary for the spent fuel. The applicant has indicated that each DSC performs the following intended functions:

- Provides criticality control of spent fuel
- Provides heat transfer for content temperature control
- Directly or indirectly maintains a pressure boundary (confinement)
- Provides radiation shielding
- Provides structural support (structural integrity)

The applicant further stated that the DSC shell assembly provides confinement of radioactive materials, encapsulates the fuel in an inert (helium) atmosphere, and provides radiological shielding (in the axial direction).

The applicant stated that the internal basket assembly, which contains a storage position for each fuel assembly, consists of an assemblage of spacer discs supported on vertical rods or an assemblage of individual tubes or a grid of plates. The basket maintains subcriticality through the geometric separation of the fuel assemblies by the DSC basket assembly and the neutron absorbing capability of the DSC materials of construction.

The applicant identified the following DSC components to be within the scope of the renewal:

- DSC Basket including: Guide Sleeves, Spacer Disks, Support Rods, Tubes, Grid Plates, Rails, Rail Inserts, and Fixed Neutron Absorbers
- DSC Shell w/ Bottom shield Plug
- Top Shield Plug
- Cover Plates (Top and Bottom)
- Siphon and Vent Port Block
- Siphon and Vent Port Cover Plates
- Grapple Ring and Grapple Ring Supports

The applicant summarized the AMR results for the DSC in the renewal application. The staff's evaluation of the AMR results is provided in this section.

The staff confirmed the accuracy of the list of DSC subcomponents provided in the renewal application with the information provided in the UFSAR. The staff confirmed the materials of construction, safety classification, and safety function of the DSC subcomponents. Based on its review, the staff concludes that the description of the DSCs are correct and acceptable.

#### 3.3.2.1 **Materials and Environments**

<u>DSC Shell and External Environment</u>

The applicant stated that the DSC shell assembly is a welded pressure vessel that consists of a cylindrical shell and the top and bottom end assemblies. The DSC cylindrical shell and the top and bottom inner cover plate assemblies form the pressure retaining confinement boundary for the spent fuel. The applicant indicated that each DSC has an internal basket assembly that maintains fuel geometry and on some designs has incorporated neutron absorber materials to maintain subcriticality.

The applicant stated that the DSC shell assembly is designed and fabricated in accordance with the provisions of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section III, Division 1, Subsection NB (ASME 1985a; 1999; 2000a; 2006), with certain code alternatives for each DSC type as described in the UFSAR. Fabrication processes including cutting, forming, machining, fitting, and aligning are performed in accordance with Subsection NB-4000 and nondestructive examination (NDE) of welds, including radiographic testing (RT), UT, and liquid penetrant test (PT) methods, is performed in accordance with ASME Code Section V and the acceptance standards of NB-5000. In addition, the applicant stated that all DSCs, except for the 24P, 24PT2, and the 52B DSC, are tested to the ANSI N14.5 leak-tight acceptance criterion of $10^{-7}$ ref $cm^3/s$ (ANSI, 1997).

The applicant reported that the DSC shell assembly subcomponents are constructed of several materials including stainless steel, carbon steel or coated carbon steel, and lead. The cylindrical shell, inner and outer top cover plates, inner and outer bottom cover plates, top and bottom shield plug forgings, grapple ring assembly, siphon and vent block, siphon and vent cover plates, support ring, lifting lugs, as well as other miscellaneous parts are constructed of stainless steel. As indicated by the applicant, all external surface of the DSC shell are stainless steel. The applicant noted that the bottom end assembly includes a stainless steel grapple ring that is used to insert and, if needed, extract the DSC from the HSM.

The applicant stated that the top end assembly consists of the top shield plug and the inner and outer cover plates. The top shield plug can be a stand-alone steel component made of carbon

steel plate or cast lead encased by carbon steel. According to the loading information provided by the applicant, the top shield plug is placed into the DSC after the SNF assemblies are loaded and before the DSC is removed from the spent fuel pool. The applicant also stated that the inner top cover plate is placed over the top shield plug and is welded to the shell to form the inner seal.

The applicant indicated that the top end assembly includes a stainless steel siphon and vent block with two penetrations into the canister cavity along with quick-connect fittings that are installed in the top of the block. The applicant stated that, after canister drying and backfilling with helium, the siphon and vent port lines are removed from the quick-connect fittings. Stainless steel plates are welded over the siphon and vent port recesses prior to installation of the outer top cover plate. The applicant stated that the stainless steel outer top cover plate is placed on top of the inner top cover plate and welded to the DSC shell to form the redundant seal required by 10 CFR 72.236(e).

The applicant stated that after loading, each DSC is positioned for storage inside an HSM and the external stainless steel surfaces of the DSC (shell, top and bottom outer cover plates, grapple assembly, and associated welds) are exposed to the HSM interior. The applicant has stated that this is a sheltered environment because the HSM prevents direct exposure to sun, wind, or precipitation, but that the internal HSM environment may contain moisture and salts or other contaminants from the external ambient air. The applicant characterized the maximum initial DSC shell assembly temperatures for normal conditions of storage for the various DSC types inside the various HSM models in Table 3.5-2 of the renewal application. The applicant determined that the maximum DSC shell temperatures are dependent on DSC type.

The applicant provided the results of computational fluid dynamics (CFD) models developed to evaluate the thermal performance of these HSMs loaded with DSCs having initial heat loads that range from 2 to 32 kW and using long-term average ambient temperature of 70 °F [21 °C]. The applicant reported that the DSC shell temperatures rapidly decrease over the initial 20 years of storage time and continue to decrease at a slower rate during the period of extended operation.

The applicant also noted that the DSC shell assembly components are exposed to neutron and gamma radiation. The renewal application includes analyses performed to evaluate the effects of neutron fluence and gamma radiation on the mechanical properties of the Standardized NUHOMS® System components. The applicant indicated these analyses were conducted using bounding sources and bounding Monte Carlo N-Particle (MCNP) models in order to envelop all the HSM and DSC configurations. As indicated by the applicant in the renewal application, the evaluation takes credit for source strength decay over an assumed storage duration of 100 years, and the energy deposition is integrated over the same period. The applicant reported that the calculated neutron fluence on the DSC shell assembly is well below the level of concern for embrittlement of the stainless steel of $1 \times 10^{18}$ neutrons/cm$^2$ (EPRI, 2007).

The applicant stated that the interaction of gamma radiation with the water, which could be present on the canister surfaces as a result of deliquescence of deposited salts, can generate radiolytic oxidizing products such as hydrogen peroxide and nitric acid. These radiolytic products could affect the corrosion process of stainless steel canister and promote radiation-induced localized corrosion including pitting corrosion, crevice corrosion, and stress corrosion cracking (SCC) of the stainless steel canister during storage. The applicant analyzed the available data and determined that the potential effect of gamma radiation on canister CISCC

during storage is unclear. As a result, gamma radiolysis of water on the canister surface and its effect on canister CISCC and crevice corrosion will be treated as an aging effect requiring management.

DSC Basket and Internal Environment

The internal basket assembly contains a storage position for each fuel assembly (FA). The basket assembly may consist of an assemblage of spacer discs supported on vertical rods (spacer disc design) or an assemblage of individual tubes or a grid of plates (tube/plate grid design).

The applicant stated that the 24P, 24P Long Cavity, 24PT2, 24PHB and 52B DSCs use the spacer disc basket design. According to the applicant, the spacer disks, spacer sleeves, and support bars are constructed from coated carbon steel. The applicant indicated the support rods, guide sleeves, and oversleeves are constructed from stainless steel. The applicant indicated that the 52B DSC design also contains borated stainless steel poison plates in the basket assembly for additional criticality control. The applicant indicated that the 61BT, 32PT, 24PTH, 61BTH, and 32PTH1, 69BTH, and 37PTH DSCs have a tube or plate grid basket design that consists of assemblies of stainless steel fuel compartments made up of individual tubes or welded plates to form a grid-like structure. The applicant indicated that fixed neutron absorber material in the basket that provides the necessary criticality control which may be borated stainless steel, boron aluminum alloy, metal matrix composite (MMC), or BORAL®. The applicant indicated that aluminum plates in the basket are used to provide the heat conduction paths from the SNF assemblies to the canister shell. The applicant stated that transition rails located between the fuel compartment structure and the cylindrical DSC shell are constructed from stainless steel plates or aluminum parts.

The applicant stated that the internal components of the DSC including the basket assembly, other shell assembly components, such as inner top and bottom cover plates, shield plugs, siphon and vent block and cover plates, and associated welds, are exposed to the inert gas (helium) environment inside the DSC cavity. In addition, the applicant reported that the average helium temperature in the DSC cavity is on the order of 300°C at the beginning of storage. The applicant stated that the helium gas temperature decreases over the period of extended operation. The applicant also stated that the helium gas pressure inside the DSC cavity under normal conditions is 1.3 psig at 20 years and atmospheric at 60 years of storage.

The applicant noted that the DSC internal components are exposed to neutron and gamma radiation. Based on bounding Monte Carlo N-Particle (MCNP) (ORNL, 2006) models the applicant reported the maximum neutron fluence in the basket assembly center fuel compartments is well below the level of concern for embrittlement of steels of $1 \times 10^{18}$ neutrons/cm$^2$ (EPRI, 2007).

NRC Staff Review of DSC Materials and Environments

The staff reviewed the applicant's description of the environment for the DSC and the DSC subcomponents. The staff evaluated the information provided by the applicant with respect to the neutron fluence interacting with the DSC components as part of the operating environment review. The staff also reviewed information provided in the UFSAR including the general description of the system, the general description of the installation in (Chapter 1), the site characteristics (Chapter 2), specific details of the DSC designs and materials of construction (Appendices E, H, K, L, N, P, T, U, Y, and Z) as well as the information provided in the renewal

application on lead system inspections of HSMs and DSCs at the Calvert Cliffs ISFSI in 2012. The staff determines that the applicant adequately defined the materials and environments for the Standardized NUHOMS System.

#### 3.3.2.2 Aging Effects/Mechanisms for the Dry Shielded Canisters During the Period of Extended Storage

The applicant evaluated the materials of construction for DSC subcomponents that are subject to further aging management review including stainless steel, carbon steel, and aluminum. For the DSC materials of construction, the applicant considered aging effects that could, if left unmanaged, cause degradation of DSC subcomponents and result in loss of the component intended function(s). The applicant's assessment of the aging effects that could cause loss of intended function(s) if left unmanaged include:

- Loss of Material
- Cracking
- Change in Material Properties

The applicant then considered the range of possible aging mechanism(s) that could lead to aging effect(s) for stainless steel, carbon steel, and aluminum components. Aging mechanisms for the DSC materials, which could lead to aging effects and ultimately a loss of component intended functions, were determined using technical publications, NUREG/CRs, and EPRI reports that are specifically identified in the following sections. These aging mechanism(s) were evaluated by the applicant to determine if the mechanism(s) could lead to an aging effect requiring management.

For loss of material, the applicant evaluated the following aging mechanism and material combinations:

- Loss of Material due to general corrosion – carbon steel
- Loss of Material due to crevice corrosion – aluminum, carbon steel and stainless steel
- Loss of Material due to pitting corrosion – aluminum, carbon steel, and stainless steel
- Loss of Material due to galvanic corrosion – dissimilar metals

For cracking the applicant evaluated the following aging mechanism and material combinations:

- Cracking due to stress corrosion cracking – stainless steel
- Cracking due to thermal fatigue – aluminum, carbon steel and stainless steel

For change in material properties the applicant evaluated the following aging mechanism and material combinations:

- Change in material properties due to elevated temperature – aluminum, carbon steel and stainless steel
- Change in material properties due to irradiation embrittlement- aluminum, carbon steel and stainless steel
- Change in material properties due to creep – aluminum

General Corrosion

The applicant indicated that general corrosion can occur when carbon steel surfaces are in contact with moist air or water. The applicant noted that the rate of general corrosion is governed by several factors, such as the moisture of the air, the salinity level of the air, the temperature of the metal surface, and the specific type of metal involved. For the DSC, the applicant stated that the carbon steel subcomponents of the DSC, including the top and bottom shield plugs and other top shield plug assembly parts, are completely enclosed within the DSC cavity which has been dried and backfilled with an inert gas (helium). The low amount of residual water remaining in the DSC cavity with an inert gas backfill is an environment that is not conducive to general corrosion. Based on the inert nature of the helium environment inside the DSC, the applicant concluded that loss of material due to general corrosion of carbon steel is not an aging effect requiring management for DSCs.

The staff have reviewed the applicant's assessment of general corrosion of carbon steel used in internal components of the DSC. The applicant concluded that general corrosion of carbon steel is not an aging mechanism requiring management. This assessment was based on the environment inside the DSC which contains very little water and is backfilled with helium. Helium is used to provide heat transfer and is also an inert gas. The staff have reviewed the applicant's description of the of the environment as well as documented assessments of atmospheric conditions on the general corrosion of carbon steel. The general corrosion rate of carbon steel in atmospheric exposures is dependent on a number of factors including humidity, time of wetness, atmospheric contaminants and oxidizing species (Fontana, 1986). Atmospheric contaminants such as chloride and sulfur species can significantly accelerate general corrosion rates; however, such species are strictly controlled in operating reactor coolant (EPRI, 1999, 2000). Because the corrosion of carbon and low alloy steels is dependent on both humidity and the oxidizing nature of the environment (Fontana, 1986), drying and backfilling the DSC cavity with an inert gas will reduce the oxidizing potential of the environment and the relative humidity inside the canister, both of which will significantly decrease the uniform corrosion rate of carbon steel. Therefore the staff finds the applicant's assessment that no aging management activity is required for general corrosion of carbon steel inside the canister to be acceptable.

Pitting and Crevice Corrosion

The applicant described pitting corrosion as localized corrosive attack in aqueous environments containing dissolved oxygen and halide contamination, such as chlorides, bromides, and hypochlorites (EPRI, 2007). As with crevice corrosion, the applicant noted that areas in which aggressive species can concentrate, that is, locations of frequent or prolonged wetting or of alternate wetting and drying, are particularly susceptible to pitting. The applicant noted that pitting corrosion is more common with passive materials, such as austenitic (300 Series) stainless steels than with nonpassive materials, such as carbon steels (EPRI, 2007).

The applicant stated that the DSC is located within the sheltered environment of the HSM interior, and DSC decay heat will heat the air preventing the accumulation or condensation of moisture inside the HSM. However, because the DSC decay heat will decrease during the period of extended operation, the applicant also stated that the presence of moist air that is potentially aggressive in contact with the DSC should still need to be considered. Therefore, the applicant concluded that pitting corrosion is an aging mechanism is potentially operative on the external surface of the stainless steel DSC shell assembly and loss of material due to pitting corrosion in stainless steel is an aging effect requiring management. The applicant stated that

the DSC steel and aluminum subcomponents located inside the DSC cavity are in an inert environment and are not subject to pitting corrosion.

The applicant stated that crevice corrosion is a form of localized corrosion that occurs in occluded spaces or crevices created by component or part connections such as lap joints, splice plates, bolt threads, under bolt heads, or points of contact between metals and non-metals. A number of factors influence crevice corrosion including electrolyte composition and flow, the geometry of the occluded region and the concentration of dissolved oxygen within the occluded region. The applicant stated that while atmospheric pollutants and contaminants are typically insufficient to promote crevice corrosion, alternating wetting and drying is particularly harmful because this leads to a concentration of atmospheric pollutants and contaminants moist air in the absence of condensation can be potentially aggressive. In addition, the applicant noted that the accumulation of aggressive species such as chloride can be influenced by concentration of these contaminants in the air. The applicant identified that more aggressive conditions may exist at ISFSI locations that are close to coastal regions, near salted roads, or in the path of cooling tower drift.

The applicant stated that the DSC is located within the sheltered environment of the HSM interior, and DSC decay heat will heat the air preventing the accumulation or condensation of moisture inside the HSM. However, because the DSC decay heat will decrease during the period of extended operation, the applicant also stated that the presence of moist air that is potentially aggressive in contact with the DSC should still need to be considered. Therefore, the applicant concluded that crevice corrosion is an aging mechanism is potentially operative on the external surface of the stainless steel DSC shell assembly and loss of material due to crevice corrosion in stainless steel is an aging effect requiring management. However, the applicant stated that the DSC steel and aluminum subcomponents located inside the DSC cavity are in an inert environment and are not subject to crevice corrosion.

The staff have reviewed the applicant's assessment of pitting and crevice corrosion of carbon steel and aluminum inside the DSC. The applicant concluded that pitting and crevice corrosion of carbon steel and aluminum are not aging mechanisms requiring management. This assessment was based on the environment inside the DSC which contains very little water after drying and is backfilled with helium, an inert gas. The staff have reviewed the applicant's description of the of the environment as well as documented assessments of conditions for pitting and crevice corrosion of carbon steel and aluminum alloys (Szklarska-Smialowska, 1986). Carbon steels are susceptible to pitting in environments containing halides and sulfur containing species such as sulfates. Aluminum alloys are susceptible to pitting and crevice corrosion in halide containing environments most notably, chlorides. Both halides and sulfur containing species are strictly controlled in operating reactor coolant (EPRI, 1999, 2000). Evacuating the canister under vacuum will significantly reduce the water content and humidity remaining inside the DSC, and backfilling the DSC cavity with an inert gas, such as helium, will reduce the oxidizing potential of the environment. Under these conditions, the formation of an electrochemical cell to support localized corrosion reactions is not possible. Therefore the staff finds the applicant's assessment that no aging management activity is required for pitting and crevice corrosion of carbon steel and aluminum inside the DSC to be acceptable.

Galvanic Corrosion

The applicant described galvanic corrosion as a form of corrosion that can occur when two or more metals of differing electrochemical potential are in electrical contact in the presence of an electrolyte. Under these conditions, an electrolytic cell is formed transmitting an electrical

current between an anode and a cathode. Loss of material occurs when ions of the metal with the lower potential, the anode, are being depleted and deposited onto the more noble metal, the cathode. The applicant also stated that corrosion rate rates increase when the more noble metal has a greater surface area than the more active metal in the presence of moisture and water (EPRI, 2007). Because an electrolyte is necessary, the applicant indicated that galvanic corrosion will not occur in a dry environment.

The applicant identified the potential for galvanic corrosion of the DSC shell and/or the stainless steel HSM rail face as a result of contact with the graphite lubricant used on the DSC support structure rail faces. The applicant stated that galvanic corrosion of the DSC shell and the DSC rail structures are possible if an electrolyte is present because the graphite lubricant is noble relative to the stainless steel materials. While the applicant stated that even though there is no reported operating experience of stainless steel degradation where a graphite lubricant film causes galvanic corrosion, loss of material due to galvanic corrosion of the DSC shell is an aging effect requiring management for stainless steel DSC shell in a sheltered environment.

The staff reviewed the applicant's assessment of the potential for galvanic corrosion as an aging effect requiring management. For internal components the applicant's assessment is based on the dry, inert environment inside the DSC. The staff have reviewed the applicant's description of the of the environment as well inside the DSC as well as the conditions necessary for corrosion of the DSC internal components that are constructed from stainless steels, carbon steels, and aluminum alloys (Fontana, 1986). Evacuating the canister under vacuum will significantly reduce the water content and humidity inside the canister, and backfilling with an inert gas, such as helium, will reduce the oxidizing potential of the environment. Under these conditions, the formation of an electrochemical cell to support localized corrosion reactions is not possible. Therefore the staff finds the applicant's assessment that no aging management activity is required for galvanic corrosion of the DSC internals to be acceptable.

The applicant identified the potential for galvanic corrosion of the DSC shell and/or the stainless steel HSM rail face as a result of contact with the graphite lubricant used on the DSC support structure rail faces. The applicant stated that galvanic corrosion of the DSC shell and the DSC rail structures are possible if an electrolyte is present because the graphite lubricant is noble relative to the stainless steel materials. The applicant stated that even though there is no reported operating experience of stainless steel degradation where a graphite lubricant film causes galvanic corrosion, loss of material due to galvanic corrosion of the DSC shell is an aging effect requiring management for stainless steel DSC shell in a sheltered environment. The staff have reviewed the materials of construction for the DSC shell and the DSC support rails with a graphite lubricant could occur due to the dissimilar electrochemical potentials for stainless steel and graphite. The staff determined that galvanic corrosion at the DSC shell/support rail interface is a potential aging mechanism requiring aging management. The staff note that no indication of preferential corrosion was observed in the DSC inspection at the Calvert Cliffs Nuclear Power Plant ISFSI (Waldrop, 2014).

<u>Stress Corrosion Cracking (SCC)</u>

The applicant stated that stress corrosion cracking (SCC) is a localized non-ductile cracking failure resulting from an unfavorable combination of applied or residual tensile stresses, material condition, and the presence of a corrosive environment. The applicant referenced the reactor operating experience, documented in Information Notice 2012-20 (NRC, 2012) showing that austenitic stainless steels under tensile stresses are known to be susceptible to SCC when exposed to chlorides in the environment. The applicant also referred to a literature survey,

which revealed failures attributed to chloride-induced SCC (CISCC) in the types of austenitic stainless steels typically used in DSS canisters when these materials are exposed to atmospheric conditions near saltwater bodies (Gorman et al., 2014). Possible sources of airborne chloride salts include seawater in coastal locations, roads treated with deicing salts, and effluents from cooling tower water which are often treated with hypochlorite biocides and other antifouling agents. The applicant indicated that these salts may enter the HSM interior, via the HSM external vents, and deposit and accumulate on the canister surface. The applicant stated that laboratory data indicates CISCC could be a concern as the canister surface temperature decreases to the level where salt will absorb moisture from the air and deliquesce.

The applicant included an evaluation of environmental conditions for CISCC of the austenitic stainless steel canister shell, an estimation of crack growth rate and penetration depth in canisters due to CISCC and an evaluation of the applicability of SCC operating experience.

Conditions for CISCC

The applicant stated that the conditions for CISCC are dependent on (1) canister surface temperature which is a function of the DSC type heat load and location on the DSC shell, (2) relative humidity near the canister surface, (3) accumulated chloride concentration on the DSC surface, and (4) residual stress.

The applicant referenced the work of Shirai et al. (2011a; b; c) and NUREG/CR-7170 (He et al., 2014) for valid measurements of the critical surface chloride concentration for CISCC. The applicant noted that surface orientation may affect accumulation rates. Because the correlations were only available for vertical orientations, the applicant used conservative assumptions of air chloride concentration and critical relative humidity to determine the time for CISCC initiation.

The applicant stated that the most important environmental parameter will be the airborne salt concentration in determining the approximate time for CISCC initiation. The applicant noted that initiation time can be slightly different with the different heat loads. The time to reach a threshold surface chloride concentration tends to shorten as the initial heat load decreases, however, there is little variation in the initiation time depending on the heat load and relative humidity (RH). The applicant identified three environmental locations where the DSCs may be susceptible to CISCC: (1) coastal locations near an open ocean (Tanaka et al., 2006; Hossain and Easa. 2011; Cole et al., 2003; Corvo et al., 1995; Castaneda et al., 2013; Feliu et al., 1999; Gustafsson and Franzen, 1996; Ambler and Bain, 1955; Meira et al., 2008), (2) locations near a major roadway that uses deicing salts (Allen and Stensland, 2006; Blomqvist and Gustafsson, 2008), and (3) locations within the effluent path of cooling towers that use chloride containing water treatment for control of biocides (Subacz, 2008; Meroney, 2005; Roffman and Vleck, 1974; NRC, 2007).

CISCC Growth Rates

The applicant evaluated the crack growth behavior due to CISCC of austenitic stainless steel DSCs, and estimated crack growth rates considering the effects of environmental conditions (e.g., canister surface temperature, humidity, and chloride concentration), stress intensity factor, and material condition (e.g., sensitization). The applicant acknowledged that it is widely accepted that the CGR can increase with temperature, stress and/or stress intensity factor (SIF), chloride concentration, humidity, and sensitization in an aqueous condition. The applicant stated that compared to the relatively large database existing for CISCC of austenitic stainless steel in the immersed state, the relevant data for the atmospheric CISCC in a dry or humid air

environment are very limited. The applicant used the available data on CISCC initiation and growth to support the frequency of inspections. The applicant stated that the bases for the inspection frequency would be refined as additional data and operational experience are obtained.

The applicant provided an evaluation of the operating experience (OE) of the external outer diameter SCC (ODSCC) of the stainless steel components at nuclear power plants (NPPs) near the ocean including the information summarized in NRC IN 2012-12 (NRC, 2012) and discusses applicability of this OE to canisters in dry storage. The applicant stated that both RH and temperature are important to determine the susceptibility of the austenitic stainless steel to CISCC in terms of crack initiation time and crack growth rate. The applicant stated that the environmental conditions affecting the canister CISCC in dry storage will be different due to relatively low RH on the hot canister surface and a sheltering effect to reduce the chloride content inside the HSM. The applicant stated that chloride concentration for the canister surface is expected to be low due to the low RH, the lack of significant temperature cycling, and benefits of the canister being sheltered.

The applicant noted that compared to the CGRs projected for the canister, the SCC events at operating plants have faster apparent crack growth rates based on the cumulative measure of crack depth over time that includes both initiation time and propagation time. The applicant stated that since the material and stress level of the nuclear power plant piping and the canister shell are generally similar, the environmental component is the primary reason the nuclear power plant piping is more susceptible to SCC than the canister shell. The applicant stated that the canister in dry storage will be exposed to a relatively low RH environment created by a hot canister surface and the sheltering effect of the HSM. The applicant stated that the operating environment inside the HSM will reduce the airborne chloride content resulting in longer time required for crack initiation and low crack growth rates compared to those observed for piping systems at operating nuclear power plants.

Effects of Temporary Attachments during Fabrication on CISCC

The applicant stated that the DSC fabrication is in accordance with ASME Code Section III, NB-4000. The applicant also stated that the welding and removal of temporary attachments results in a shallow heat affected zone and that detrimental through thickness microstructural alteration does not occur. Because the microstructural effects are limited and because the welding and removal of temporary attachments is done per the provisions of the ASME Code that includes required examinations, the applicant determined there is a low risk that SCC will occur at these locations. Therefore, the applicant concluded that it is not necessary to search for temporary attachment locations specifically for inspection during the period of extended operation.

The staff reviewed the applicant's assessment of temporary attachments and the relevant portions of the ASME B&PV code. The staff determined that the process of welding and removing temporary fixtures to the DSC shell is unlikely to result in through wall tensile residual stresses necessary for the initiation and propagation of SCC. Therefore the staff concludes that the applicant's assessment of temporary attachments is adequate.

Thermal Fatigue

The applicant stated that thermal fatigue is the progressive and localized structural damage that occurs when a material is subjected to cyclic loading associated with thermal cycling. The applicant stated that the only source of potential thermal fatigue of the DSC is ambient seasonal

and daily temperature fluctuation; however, because of the large mass of the loaded DSC, the DSC does not experience the full amplitude of ambient temperature cycles, and a gradual, long-term temperature decrease occurs during the course of storage. In addition the applicant stated that the magnitude of the thermal effects from seasonal and daily variations in ambient conditions are reduced by the thermal mass of the DSC, the DSC contents and the HSM. The applicant provided an evaluation of the DSC for pressure and temperature fluctuations in accordance with the provisions of NB-3222.4(d) of the ASME Code (ASME, 1985a, 1999, 2000a, 2006). The applicant stated that as provided by NB-3222.4(d) of the ASME Code, fatigue effects need not be specifically evaluated provided the six criteria in NB-3222.4(d) are met. The applicant stated that the evaluation is performed considering a 100-year service life using maximum initial DSC pressures (bounding for all DSCs) and temperatures at the beginning of storage, and assumes that the maximum pressures and temperatures do not decrease over time, but may fluctuate due to ambient condition changes. Because the criteria of NB-3222.4(d) are met, the applicant concluded that cracking due to thermal fatigue is not an aging effect requiring management for DSC subcomponents.

The staff reviewed the applicant's calculations and the criteria from the ASME Code, NB-322.4(d) and finds that because the criteria are met, thermal fatigue is not an aging effect requiring management for DSC subcomponents. The staff's review of the of the TLAA on the thermal fatigue of DSC subcomponents is documented in Section 3.4.1 of this safety evaluation report.

Elevated Temperature Exposure

The applicant stated that the maximum DSC internals temperatures are limited by the cladding temperature limit of 400 °C [752 °F] at the beginning of storage per ISG-11 Rev 3 (NRC, 2003). In addition the applicant stated that the DSC subcomponent initial material temperatures for normal conditions of storage were shown to be within the temperature limits allowed by the ASME Code or were evaluated to show they can perform their safety function.

The applicant analyzed the potential for thermally induced degradation of the DSC welds by embrittlement of the delta ferrite phase. The applicant cited the results published in NUREG/CR-6428 that showed full embrittlement of the delta ferrite phase which is typically 5 to 12 percent volume percent of the weld metal does not lead to significant embrittlement of the entire weld as confirmed by measured values of fracture toughness. For the DSC shell, the applicant stated that the maximum temperature of the DSC shell is approximately 250 °C [482 °F] at the beginning of storage. The applicant cited NUREG-1801 Revision 2 (NRC, 2010d), which states that austenitic steels with service temperature below 250 °C [482 °F] are not susceptible to thermal aging embrittlement. The applicant also cited the results the spent fuel storage demonstration tests using a Castor V/21 reported in NUREG/CR-6745 (Bare and Torgerson, 2001) which showed no evidence of stainless steel basket weld degradation due to thermal aging after 14 years with fuel in the dry storage cask.

The applicant stated that stress evaluations for the aluminum components of the basket assembly are based on conservative mechanical properties to account for any changes in strength that may occur under exposure to elevated temperatures.

The applicant also stated that DSC subcomponent materials temperatures will significantly decrease over the period of extended operation as the decay heat of the spent fuel in the DSC decreases. Therefore, the applicant concluded that change in DSC material properties due to elevated temperature is not an aging effect requiring management for DSC subcomponents.

The staff reviewed the applicant's assessment of the potential for change in material properties as a result of exposure to elevated temperatures. The applicant concluded that the exposure temperatures are either within the ASME Code allowable temperature ranges or were evaluated to show that the materials can perform their safety function at the operating temperatures. The staff reviewed relevant information on the effects of thermal exposures on stainless steel welds including the work of Gavenda et al. (1996). This information indicated that welded stainless steels can undergo thermal embrittlement of the delta ferrite phase at temperatures above 335°C [635°F]. The staff determined that because of the operating temperatures, thermal aging of stainless steel welds would only occur for internal components such as the welds in the stainless steel basket. The staff reviewed the information provided and determined that because the stainless steel welds typically contain less than 12 percent delta ferrite, thermal aging of the welds would only result in marginal decreased in fracture toughness. For the DSC shell, the shell temperatures are too low to be adversely affected during storage.

Neutron Fluence and Gamma Radiation

The applicant provided analyses performed to evaluate the effects of neutron fluence and gamma radiation on the mechanical properties of the Standardized NUHOMS® System components. The applicant reported that these analyses used bounding sources, and that bounding MCNP models are used in order to envelop all the possible HSM and DSC configurations/combinations. The applicant stated that the evaluation takes credit for source strength decay over an assumed storage duration of 100 years, and that the energy deposition is integrated over the same period. The applicant calculated maximum integrated neutron fluence for the fuel compartment at the center of the basket and the maximum neutron fluence on the DSC shell assembly. Because these calculated values are well below the level of concern for embrittlement of stainless steel, reported to be $1 \times 10^{18}$ n/cm$^2$ (EPRI, 2007), the applicant concluded that change in material properties due to irradiation embrittlement is not an aging effect requiring management for DSC subcomponents.

The applicant used General Electric (GE) 7×7 fuel with 0.198 MTU for BWR source term and Babcock and Wilcox (B&W) 15×15 fuel with 0.490 MTU for the PWR source term calculations. The SAS2 module in SCALE is used to generate the source terms using the active fuel input from 69BTH and 32PTH.

The applicant used bounding parameters for enrichment, burnup and minimum cooling time for a pressurized water reactor (PWR) fuel, which is bounding for all PWR DSCs. The heat load used in the analysis of 124.8 kW/DSC is almost three times of the maximum heat load of all PWR DSCs (40.8 kW/DSC for 24PTH and 32PTH1). The neutron source per DSC (5.4496E+10 n/(s*DSC)) used in the analysis is 15 percent higher than the maximum design bases neutron source per DSC (4.74E+10 n/(s*DSC) from the 32PTH1) of all PWR DSC options. The applicant concluded that the conservative source terms used in the analysis provide reasonable assurance that the results based on the 32PTH1 DSC are bounding for all PWR DSCs.

The applicant analyzed the BWR DSCs the same way as it did for the PWR and concluded that the source terms used in the analysis provide reasonable assurance that the results based on the 69BTH DSC are bounding for all BWR DSCs.

Staff independently verified the applicant's conclusion by calculating neutron source term using Origin-ARP module of Scale 6.1 and concluded that 32 PTH1 DSC is bounding for all PWS DSCs and 69 BTH DSC is bounding for all BWR DSCs.

The applicant developed two MCNP bounding models, one for storing 32PTH1 and one for storing 69BTH including shell assembly and the internal assembly based on the shielding analyses to store in the HSM internal cavity. In these models there is no gap between DSC and HSM concrete walls and roof surface to maximize the neutron fluence and gamma energy deposition. The heat shields for thermal protection of HSM are not modeled.

The HSM is modeled as a 1 foot thick cylindrical concrete shell. The front and back walls are modeled also as 1 foot thick concrete.

These MCNP models are used to calculate the neutron fluence levels at the center stainless steel basket compartment and at the DSC shell. Both neutron fluence and gamma exposure levels are calculated at the concrete inside the HSM cavity.

The staff determined the calculated values are well below the level of concern for embrittlement of stainless steel, reported to be $1\times10^{18}$ n/cm$^2$ for both 32 PTH1 and 69 BTH DSCs.

The staff independently developed MCNP models to verify the applicant's conclusions. The results from staff evaluations match with the applicant's results; therefore, the staff finds the neutron fluence and gamma radiation models to be bounding and acceptable.

Creep of Metal Components

The applicant stated that creep is a time-dependent strain (deformation) in metals occurring under constant load at constant temperature. The applicant stated that an increase in either stress or temperature accelerates creep and if stress or temperature is increased beyond certain levels, the increased deformation can eventually result in failure.

The applicant stated that metallic materials are generally considered to be subject to creep under conditions of extended exposure to stress and temperature in excess of a homologous temperature of 0.4Tm, where Tm is melting point in degree Kelvin. For aluminum alloys, this translates to a temperature of approximately 100 °C [212 °F]. The applicant stated that the material of interest for assessment of degradation due to creep is the aluminum parts in the basket assembly. Certain basket subcomponents in the 61BTH, 32PTH1, 24PTH, 32PT, 69BTH and 37PTH baskets are also fabricated from aluminum alloys.

The applicant stated that except for the aluminum inserts, which carry no load other than their own self-weight, the aluminum components are in a compressive state of stress. In addition, the applicant stated that these components are restrained by other stainless steel compartments. Because these aluminum components are confined, the applicant stated that their deformation is contained.

The applicant stated that a creep evaluation of the above-basket aluminum components was performed in response to CoC 1004 Amendment 10 Request for Additional Information (RAI) 9-11 (AREVA TN, 2007). The applicant conducted the evaluation using the allowable stress criteria based on the ASME Code, which states in Section II, Part D, Appendix 1 (1998), 1-100 4(b): "At temperatures in the range where creep and stress rupture govern the selection of stresses, the maximum allowable stress value for all materials is established by the Committee not to exceed the lowest of the following:

(1) 100% of the average stress to produce a creep rate of 0.01%/1000 hour;
(2) 67% of the average stress to cause rupture at the end of 100,000 hours;

(3) 80% of the minimum stress to cause rupture at the end of 100,000 hours."

The applicant stated that because the aluminum components are in compression, creep rupture is not of concern and hence the creep rate criteria in (1) is governing. The applicant stated that because the maximum stresses in storage are small relative to the calculated the allowable stress, creep is not a significant consideration in the design of the NUHOMS® baskets.

The staff reviewed the applicant's statement that the aluminum components of the basket assembly perform a thermal function and are not credited for a structural function and that the stress evaluations for the basket rails in the UFSAR are based on conservative mechanical properties for aluminum alloys. The staff determined that the use of conservative mechanical properties is adequate to account for the changes in strength that may occur under when the aluminum alloys are exposed to elevated temperatures.

Neutron Absorber Thermal-Mechanical Degradation and Boron Depletion

The applicant stated that the neutron absorber used for criticality control in the 32PT DSC basket may consist of borated aluminum, or a boron carbide/aluminum MMC. The applicant stated that if required, depending on SFA design and initial enrichment, poison rod assemblies (PRAs) which are inserted in the guide tubes of certain assemblies in the basket, are left in place following the completion of the DSC draining and drying operations, may also be used for criticality control. For the 61BT, 24PTH, 61BTH, 32PTH1, 69BTH, and 37PTH DSCs, the applicant stated that the neutron absorber used for criticality control in the DSC basket may consist of borated aluminum, boron carbide/aluminum MMC, or BORAL®.

The applicant stated that the safety analyses do not rely upon the mechanical strength of the neutron absorber materials. In addition, the applicant stated that the performance of the neutron absorber's design function is ensured only by the presence of B-10 and the uniformity of its distribution, which are verified with testing requirements specific to each material. The applicant indicated that the radiation and temperature environment in the DSS is not sufficiently severe to damage these metallic/ceramic materials.

In the renewal application, the applicant provided a bounding evaluation of B-10 depletion of the neutron absorber plates based on the maximum neutron source terms and minimum B-10 areal density of all the CoC 1004 DSCs. The TLAA determines the amount of B-10 depleted in the poison plates due to neutron irradiation. The applicant stated that the evaluation shows that 99.999% of B-10 remains in the poison plates after the period of applied CoC renewal. Based on the result of this TLAA that demonstrates that the ability of the poison plates and PRAs to maintain sub-criticality remains unaffected over the desired duration of storage, the applicant concluded that boron depletion is not an aging effect requiring management.

The results of the applicant's TLAA for Dry Shielded Canister Poison Plates Boron Depletion are included in the renewal application. The applicant indicated that for this evaluation, an MCNP5 model of a bounding fuel assembly and basket compartment was used. From the UFSAR, the B&W 15x15 Mark B is identified as the bounding fuel that produces the maximum neutron source per assembly and the basket used in the analyses was that for the 24PTH DSC. The applicant noted that although the boron content in the 24PTH is not the lowest boron content basket among the DSCs in the Standardized NUHOMS® System, using the 24PTH DSC as the bounding configuration is justified based on the combined consideration of neutron strength and boron content because the DSCs with lower boron contents also have fuel assemblies with about 10 times lower neutron source strengths. The applicant performed a sensitivity

calculation using the maximum neutron source and a poison plate with half as much Boron-10 as in the 24PTH DSC slightly less than the boron content in the 52B DSC, which has the lowest boron content, a bounding model, with maximum source strength and minimum Boron-10 content of all DSCs. The results demonstrated that minimal depletion occurs with the irradiation of poison plates. The applicant also analyzed the other DSCs and obtained similar results, and concluded that only less than 0.001% amount of B-10 is depleted.

The staff verified by independent analysis than negligible amount of B-10 is depleted in the poison plates due to neutron irradiation and concluded that the applicant's results of less than 0.001% of B-10 is depleted is correct.

Coating Degradation

The applicant stated that there are no coatings applied to the DSC shell assembly external surfaces exposed to the HSM sheltered environment. The applicant confirmed that protective coatings such as aluminum thermal spray and electroless nickel coatings are used to mitigate corrosion of subcomponents that are part of the DSC shell assembly and some basket assembly subcomponents. The applicant stated that the coatings limit rust contamination of the pool water, but do not perform any important to safety dry storage function. In addition, the applicant stated that after fuel loading the DSC is sealed, dried, and backfilled with helium, and the coated carbon steel components inside the DSC are maintained in an inert environment and not subject to corrosion. Based on the intended purpose of the coatings and the inert environment during storage, the applicant concluded that no aging management is required for the aluminum thermal spray or electroless nickel coatings within the DSC.

The staff reviewed the design of the DSC with respect to the use of coatings and determined that all components that utilize protective coatings are on the interior of the DSC and are not exposed to environmental conditions where coating degradation would need to be considered. The staff determines that the intended purpose of the coatings is consistent with their application and none of the coatings used on DSC components require aging management.

#### 3.3.2.3 Evaluation of the Proposed Aging Management Activities

The applicant provided time limited aging analyses (TLAA) for the following potential aging effects for the DSC:

- Fatigue Evaluation of the Dry Shielded Canisters
- Dry Shielded Canister Poison Plate Boron Depletion Evaluation
- Evaluation of Neutron Fluence and Gamma Radiation on Storage System Structural Materials
- Confinement Evaluation of the 24P and 52B Non-Leak tight Dry Shielded Canisters

The staff reviewed the CoC holder's assessment of the aging effects and mechanisms for the DSC during the period of extended operation. For identification of credible aging effects, the staff utilized information contained in NUREG-1801 (NRC, 2010a), the operational experience such as those summarized in IN 2012-20 (NRC, 2012), results of NRC sponsored research (He et al., 2014) and recent industry sponsored reviews (Gorman et al., 2014; Fuhr et al., 2013). Where needed, the staff also utilized material properties from the ASME Code (ASME, 2013) as well as specific research and reports relating environmental parameters on aging effects and aging mechanisms applicable to the materials utilized in the DSCs.

The staff reviewed the applicant's assessment and determined that analysis of these potential aging effects using a TLAA is appropriate. The fatigue evaluation of the dry shielded canisters is performed according to the requirements of the ASME Code, NB-3222.4(d). The staff review of the applicant's TLAA on fatigue of the DSC materials is included in SER section 3.4.1. The boron depletion analysis is based upon a bounding analysis of the highest neutron fluence. The staff review of the applicant's boron depletion TLAA is included in SER section 3.4.4. The applicant's evaluation of neutron fluence and gamma radiation on storage system structural materials relies on knowledge of the radiation effects on material and is supported by an analysis of the neutron fluence and gamma radiation. The staff review of the applicant's neutron fluence and gamma radiation effects TLAA is included in SER section 3.4.5. The staff review of the confinement evaluation of 24P and 52B non-leaktight DSCs is included in SER section 3.4.6

In addition, the applicant provided additional analysis in the form of a TLAA to address:

- Bounding Evaluation of Dry Shielded Canister with Reduced Shell Thickness Due to Chloride Induced Stress Corrosion Cracking under Normal and Off Normal Conditions of Storage during Renewal Period
- Defense-in-Depth Structural Evaluation of Dry Shielded Canister Confinement and Retrievability Assuming High Burnup Fuel Cladding Failure during Period of Extended Operation
- Defense-in-Depth Dose Assessment Assuming Breach of Confinement during Period of Extended Operation

The staff review of the applicant's evaluation for the bounding evaluation of a DSC with reduced shell thickness due to CISCC is provided in SER section 3.4.13. Staff review of the defense in depth structural evaluation of a DSC confinement and retrievability assuming high burnup cladding failure is provided in SER section 3.4.12. Staff review of the defense-in-depth dose assessment of DSC internal pressures assuming high-burnup cladding failures is provided in SER section 3.4.11

The applicant has also developed two aging management programs to address potential aging effects and mechanisms:

- DSC External Surfaces Aging Management Program
- DSC Aging Management Program for the Effects of CISCC

The applicant has concluded that the welded stainless steel DSC shell exposed to the sheltered environment inside the HSM may be susceptible to pitting corrosion, crevice corrosion and stress corrosion cracking and these are aging effects that require management in the period of extended operation. The staff finds that the applicant's assessment is consistent with reactor operational experience such as those summarized in IN 2012-20 (NRC, 2012) and the results of NRC sponsored research (He et al., 2014) showing that pitting corrosion, crevice corrosion and stress corrosion cracking of stainless steel components have been observed when exposed to atmospheric conditions where chloride containing species are present. The applicant's assessment is also consistent with recent industry sponsored reviews of aging mechanisms for the welded stainless steel (Gorman et al., 2014; Fuhr et al., 2013). Therefore the staff finds the applicant's assessment that aging management is required to manage the effects of crevice corrosion, pitting corrosion and stress corrosion cracking are activity is required for general corrosion of carbon steel inside the canister to be acceptable. The staff review of these AMPs is located in SER section 3.6.

### 3.3.3 Horizontal Storage Modules (HSM)

The renewal application includes HSM models 80, 102, 152, and 202, HSM-H, and HSM-HS. The differences between these modules were summarized in Section 3.6.1 of the renewal application. The applicant identified the relevant design drawings in the renewal application.

The applicant described the HSM as a low profile, modular, reinforced concrete structure. Its intended functions are to provide a means for passively removing spent fuel decay heat, provide structural support and environmental protection to the loaded DSC, and provide radiation shielding protection. The applicant stated that heat removal in the HSM is achieved by a combination of radiation, conduction, and convection. Ambient air enters the HSM through ventilation inlet openings located in the lower region of the front or side walls and circulates around the DSC. Air exits through outlet openings in the top regions of the HSM walls. Thermal monitoring or visual inspections are used to indicate HSM performance or a blocked vent condition.

Structural support of the loaded DSC is provided by a structural steel frame structure (HSM model 80 and model 102) anchored to the floor slab and walls of the HSM, or a structural steel rail assembly (HSM models HSM-H, -152, -202, and HSM-HS). Environmental protection and radiation shielding is provided by massive thick side reinforced concrete walls and roof of the HSM, supplemented by thick reinforced concrete wall units attached at the ends of the array and at the rear walls of the HSM if the array is of single row configuration.

The applicant stated that the HSMs are designed in accordance with ACI-349 and constructed in accordance with the ACI-318 (ACI, 1983; ACI, 1995). The applicant further stated that the HSM models 80, 102, 152, and 202 are designed in accordance with the requirements of ACI 349-85 (ACI, 1985), and that the HSM-H and HSM-HS models are designed in accordance with the requirements of ACI 349-97 (ACI, 1997).

The applicant stated that the HSM subcomponents include (but are not limited to) the HSM reinforced concrete walls, roof, and end/rear shield walls; DSC steel structure support assembly; HSM accessories (DSC seismic retainer, cask/HSM restraint system, heat shield panels, shielded door assemblies and door supports); associated attachment/installation hardware (tie rods, bolts, nuts, washers, embedment assemblies, mechanical splices); ventilation inlet vent openings and bird screens, ventilation outlet vent openings and bird screens, and outlet vent reinforced concrete covers.

The applicant stated that the HSMs are installed on a load bearing foundation, which consists of a reinforced concrete pad on a subgrade engineered to be suitable to support the loads. The applicant further stated that there are no structural connections or means to transfer shear between the HSM base unit module and the foundation slab. The concrete pad is classified as not important-to-safety (NITS).

The staff reviewed the descriptions of the HSM models in the renewal application and verified the accuracy with the appropriate sections and relevant design drawings in the UFSAR. The design drawings verify that the HSM models 80, 102, 152, and 202 are designed in accordance with the requirements of ACI 349-85 (ACI, 1985), and that the HSM-H and HSM-HS models are designed in accordance with the requirements of ACI 349-97 (ACI, 1997). The drawings also state that the structural steel for the prefabricated HSM is designed and constructed in accordance with the AISC “Specification for Structural Steel Buildings”, latest edition. Based on

its review, the staff concludes that the applicant adequately described the various HSM structures.

#### 3.3.3.1 **Materials and Environments**

Materials

The applicant defined the materials of construction for the subcomponents of the HSM in the renewal application (reproduced below).

**Table 3.3-9. Materials in HSM Subcomponents**

| HSM Subcomponents | Material |
|---|---|
| HSM walls, roof, floor (HSM Model 80/102 only); end shield walls, rear shield walls | Reinforced concrete |
| DSC support structure: structural beams and frames | Carbon steel<br><br>Stainless steel (HSM Model 152 only) |
| DSC support rail plate | Nitronic® 60 stainless steel |
| Shielded access door encased concrete core (HSM Model 80) | Plain Concrete |
| Shielded access door reinforced concrete (HSM Models 102, 202, and 152, HSM-H, and HSM-HS) | Reinforced concrete |
| Shielded access door: door steel plates | Carbon steel |
| Axial retainer assembly: tube steel, plate embedment, DSC axial retainer | Carbon steel<br><br>Stainless steel (HSM Model<br><br>152 only) |
| HSM Heat Shields | Carbon steel<br><br>Stainless steel<br><br>Aluminum |
| HSM cask docking ring | Carbon steel |
| HSM roof attachment angles and stiffener plates | Carbon steel |

| Table 3.3-9. Materials in HSM Subcomponents | |
|---|---|
| HSM air inlet and outlet liners and dose reduction hardware | Carbon steel |
| Support frames for HSM inlet and outlet vent bird screens | Stainless steel |
| Anchorages | Carbon steel<br><br>Stainless steel |
| End and rear shield walls-to-module connecting hardware | Carbon steel |
| HSM-HS module-to-module seismic ties and connecting rods | Carbon steel |

Environments

The applicant performed a review of the information presented in the UFSAR, to assess the environmental conditions to which the SSCs are normally exposed. The applicant stated that HSMs are located outdoors; thus, the exterior surfaces of the HSM are exposed to all weather conditions, including insolation, wind, rain, snow, plant-specific ambient temperature, humidity, and airborne contamination. The applicant further stated that the design configuration of a NUHOMS® ISFSI consisting of an array of individual HSM storage modules provides an effective means of protection against extreme seasonal weather conditions, including heavy precipitation, drifting snow, ice floes, lightning strikes, strong winds, wind driven missiles, and blowing dust. ISFSIs may be located in environments that could be classified as corrosive based on plant operating experience, which has shown exposure to aggressive species (such as saltwater atmosphere, sulfur dioxide) in industrial areas or areas near the seashore. The applicant defined the environment for exterior surfaces of the HSM as external (yard and outdoor).

The applicant stated that the temperature environment for the HSMs is bounded by the temperatures due to design bases heat load of a particular DSC type, as shown in the renewal application, at the minimum and maximum normal ambient temperatures (for both normal and off-normal conditions of storage, as defined in the renewal application).

The applicant stated that subcomponents interior to the HSM (interior side of the HSM walls, HSM steel, and DSC external shell assembly components) are considered to be in a sheltered environment because they are located in a protected environment in uncontrolled air with no direct exposure to sun, wind, or precipitation. However, the applicant clarified that a sheltered environment may contain moisture and salts or other contaminants from the external ambient air. The temperature inside the HSM depends on the ambient air temperature and the heat load of the loaded canister; the applicant defined bounding values per the conditions in the UFSAR. The applicant summarized the maximum temperatures in the HSM concrete for normal and off-normal conditions for the various DSC types with their design bases heat loads inside the corresponding HSMs. The bounding maximum predicted temperatures in the HSM concrete at

the beginning of storage for normal and off-normal conditions are defined in the UFSAR. These initial temperatures are below the design temperature limit of 300 °F identified in Section 3.5.1.2 (i)(2)(b) of NUREG-1536, Revision 1 (NRC, 2010a). NUREG-1536, Revision 1, states that if concrete temperatures of general or local areas exceed 93 °C (200 °F) but would not exceed 149 °C (300 °F), no test to prove the capability for elevated temperatures or reduction of concrete strength are required if Type II cement is used and aggregates are selected which are acceptable for concrete in this temperature range. The temperatures of the concrete are expected to decrease over the periods of extended operation.

The applicant also stated that the subcomponents in the sheltered environment of the HSM are exposed to neutron and gamma radiation to a lesser extent than those of the interior cavity of the DSC. The applicant submitted a time-limited aging analysis, which calculated a maximum bounding neutron fluence and maximum gamma radiation on the HSM concrete walls. The applicant concluded that these values are below the levels affecting compressive and tensile strength for concrete and for potential embrittlement of carbon steel reinforcing bars. The radiation sources are expected to decrease over the period of extended operation.

The applicant also defined an embedded or encased (sealed) environment, which include the rebar and anchorage subcomponents embedded in the HSM concrete. The applicant further stated that embedded or encased environments are exposed to radiation.

The applicant stated that no component of the Standardized NUHOMS® System is exposed to a soil environment. Therefore, this environment does not apply to this renewal. The applicant clarified that, at the ISFSI, the HSMs are installed on a reinforced concrete basemat, constructed on compacted, engineered fill.

The staff reviewed the descriptions of the HSM models in the renewal application and verified the accuracy of the materials of construction with the appropriate sections and relevant design drawings in the UFSAR. The design drawings verify that the HSM models 80, 102, 152, and 202 are designed in accordance with the requirements of ACI 349-85 (ACI, 1985), and that the HSM-H and HSM-HS models are designed in accordance with the requirements of ACI 349-97 (ACI, 1997). The drawings also state that the structural steel for the prefabricated HSM is designed and constructed in accordance with the AISC “Specification for Structural Steel Buildings”, latest edition.

The staff also reviewed the UFSAR, which defines the environmental characteristics for ISFSIs sited within the controlled area under the jurisdiction of the current 10 CFR 50 operating license for the plant, including the discussions on geography, meteorology, hydrology, seismology and geology. The UFSAR further identifies the HSM design loadings, including normal and off-normal operating ambient temperatures. Section 8.1.1.C states that the normal operating seasonal average daily ambient temperature fluctuates from 0°F minimum (winter) to 100°F maximum (summer) and is conservatively assumed to occur for a sufficient duration to establish steady state conditions for the transfer cask, DSC, and HSM. Section 8.1.1.C further states that these minimum and maximum steady-state long-term ambient design temperatures envelop the 24-hour average seasonal ambient temperature at any location within the contiguous United States. Section 8.1.1.C further states that the long-term average normal ambient temperature for the 50-year design life of the system is assumed to be 70 °F. Section 8.1.1.C further states that the base case average ambient temperature bounds practically all reactor sites within the United States.

The staff further reviewed the design bases in the UFSAR to identify ranges of ambient temperatures and maximum temperatures for the Horizontal Storage Module and respective subcomponents. The UFSAR clarifies the differences between the HSM model 80 and HSM model 102. It further states that the heat transfer capability of both HSM models are equivalent. The maximum calculated temperatures for the various structural sections of the HSM-80 and HSM-120 models for normal operating conditions are summarized in the UFSAR. A more detailed tabulation of the HSM thermal results used for the structural design of the HSM-80 and HSM-120 models is identified in the UFSAR. The UFSAR provides the results of the thermal analysis for the HSM support structure for both DSC-24P and 52B systems.

The UFSAR provides a summary of the maximum concrete temperatures for the HSM model 152 and HSM model 202, respectively. The maximum component temperatures for the normal and off-normal cases of the HSM-H system are listed in the UFSAR. The UFSAR states that the HSM model HS is a modification of the HSM model H design, which allows use of the use of the system in locations where higher seismic levels exist. The modifications of the HSM-HS model do not affect the temperatures observed by the HSM subcomponents relative to the HSM model H.

The staff evaluated the design drawings to determine whether the service environments of all subcomponents were adequately described as either external, sheltered, or embedded. The staff further verified the bounding maximum temperatures in the HSM concrete at the beginning of storage, per the UFSAR, and confirmed that these temperatures did not exceed those identified in NUREG-1536, Revision 1 (NRC, 2010a), which would require additional testing. The staff agrees with the applicant's conclusion that these temperatures are expected to decrease over the period of extended operation because of the decaying heat of the loaded fuel. In addition, the staff agrees with the applicant's conclusion that neutron and gamma-induced degradation of the concrete and steel subcomponents of the HSM are not credible, because the expected cumulative neutron and gamma doses for the extended period of operation are below the levels affecting the compressive and tensile strength of concrete and below the levels that could result in potential embrittlement of steel subcomponents. Based on its review of the design bases in the UFSAR, the staff concludes that the applicant adequately identified the materials of construction and service environment of the horizontal storage modules.

#### 3.3.3.2 **Aging Effects/Mechanisms for the Horizontal Storage Modules During the Period of Extended Storage**

##### 3.3.3.2.1 Reinforced Concrete Structure

The applicant listed potential aging effects/mechanisms for the HSM above-grade reinforced concrete structure in the renewal application. The applicant described three aging effects that could lead to a loss of intended functions of these subcomponents. First, the aging effect of loss of material was described as manifesting as scaling, spalling, pitting, and erosion, as described in ACI 201.1R (ACI, 2008a). Second, the aging effect of cracking was described as manifesting in concrete components as a complete or incomplete separation of the concrete in two or more parts, as depicted in ACI 201.1R (ACI, 2008a). Third, the aging effect of change in materials properties was described to manifest in concrete components as increased permeability, increased porosity, reduction in pH, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength.

The applicant further identified the various aging mechanisms that could lead to these described aging effects, and justified their applicability to the extended period of operation.

The staff reviewed these degradation mechanisms against those described in the example AMP for reinforced concrete structures in Appendix B of NUREG-1927, Revision 1. The staff reviewed the supplemental analyses in the renewal application in Section 3.4 of this SER. The staff finds that the applicant's aging management review has adequately identified credible aging effects and mechanisms for reinforced concrete based on the guidance Table B-2 of NUREG-1927, Revision 1, and technical bases cited therein. Those mechanisms not addressed in Table B-2 of NUREG-1927, Revision 1, are reviewed below.

Cracking, Loss of Material Due to Shrinkage and Creep

The applicant stated that, according to ACI 209R-92 (ACI, 2008b), most of the shrinkage in the reinforced concrete has already occurred (91% in the first year, 98% in five years and 100% in 20 years). Since 20 years will have occurred by the end of the initial approved storage term, the applicant concluded that concrete shrinkage is not a credible aging mechanism during the period of extended operation.

The applicant clarified that creep-induced concrete cracks are typically not large enough to result in concrete deterioration or in exposure of the reinforcing steel to environmental stressors, and that cracks of this magnitude do not reduce the concrete's compressive strength. The applicant further stated that creep is significant when new concrete is subjected to load; however, creep decreases exponentially with time. According to ACI 209R-92 (ACI, 2008b). 78% of creep occurs within the first few years, 93% within 10 years, 95% within 20 years and 96% within 30 years. Because of this, the applicant concluded that concrete creep is not a credible aging mechanism.

The staff reviewed the ACI 209R-92 (ACI, 2008b) and notes that the values stated by the applicant are recommended values based on 6-inch thick concrete. Table 2.5.5.1 of ACI 209R-92 (ACI, 2008b) provides correction factors for concrete thicknesses from 2 inches to 15 inches. Based on Table 2.5.5.1, the ultimate values of shrinkage and creep stated by the applicant will be reduced by 82% and 86% respectively for a thickness of 12 inches and 74% and 80% for a thickness of 15 inches. Because the minimum thickness of the HSMs is approximately 12 inches, the staff determines that shrinkage and creep are not credible aging mechanisms.

Change in material properties due to irradiation (embrittlement)

Per the discussion in Section 3.4.5 of this SER, the applicant calculated a cumulative neutron fluence on the inside surface of the HSM concrete walls during the period of extended operation, which was shown to remain below the critical neutron fluence limits, as defined in Table B-2 of NUREG-1927, Revision 1. Therefore, the staff concludes that change in mechanical properties due to neutron and gamma irradiation (embrittlement) is not a credible aging effect requiring management.

3.3.3.2.2 Steel and Aluminum Subcomponents

The applicant stated that metallic subcomponents within the HSM are stainless steel, aluminum, or carbon steel. The applicant stated that carbon steel subcomponents are coated with inorganic coatings or are galvanized for corrosion protection. Although the applicant did not assume credit for the coatings in carbon steel subcomponents, the application provided an AMR assuming coating failure to determine if it could adversely affect the safety function of a safety-related SSC.

The applicant referenced NUHOMS® HSM fabrication specifications to define the initial surface conditions of these subcomponents.

The applicant stated that the amount of applied coating within an HSM is limited. Therefore, the applicant concluded that coating failure (e.g., blistering, cracking, flaking, peeling) will not adversely affect the thermal performance of the HSM and will not prevent the HSM or the DSC from satisfactorily accomplishing its intended functions.

Although a failure of the coating does not have an adverse effect on the intended functions of the HSM, the applicant elected to include inspection attributes the HSM AMP for External and Internal Surfaces to manage loss of coating integrity.

The applicant stated that the sliding surfaces of the DSC support rails of the HSM are coated with a dry film lubricant, a not important-to-safety (NITS) component. Since the lubricant is used only for reducing friction while sliding a DSC along the support rail, once the DSC is in place within the HSM, the lubricant performs no intended function during storage of the DSC. Thus, the applicant concluded that lubricant failure would not prevent the HSM or the DSC from satisfactorily accomplishing its intended functions, and that degradation of the dry lubricant does not require aging management.

The applicant stated that additional considerations for coastal environments are also noted in a fabrication technical specification 4.2.1 in order to improve corrosion resistance of loadbearing carbon steel DSC support structure rail components of the HSMs.

The applicant listed potential aging effects/mechanisms for the carbon steel, stainless steel, and aluminum HSM subcomponents in Sections 3.6.4.3 and 3.6.4.4 of the renewal application. The applicant described three aging effects that could lead to a loss of intended functions of these subcomponents: (1) loss of material, (2) cracking, and (3) change in materials properties. The applicant further identified the aging mechanisms that could lead to the previously described aging effects, and justified their applicability to the extended period of operation.

The staff finds that the approved design-bases adequately mitigates corrosion of steel and aluminum subcomponents in the HSM through the use of NITS coatings/galvanizing. In addition, the staff considers that the applicant's assumption that coating degradation could occur to be adequate.

The following discussions provide the staff's assessment of the applicant's conclusion pertaining aging effects of steel and aluminum subcomponents:

<u>Loss of Material Due to General Corrosion – Carbon Steel</u>

The applicant stated that the atmosphere inside the HSM modules will be dependent on the site location and environment, but is typically benign in terms of corrosion. The applicant further stated that, although DSC decay heat will heat the air preventing the accumulation or condensation of moisture inside the HSM, the decay heat will decline during the period of extended operation. Thus, the presence of moist air cannot be ruled out. Therefore, the applicant concluded that loss of material due to general corrosion is an aging effect requiring management for carbon and low alloy steel in outdoor and sheltered environments. The applicant stated that stainless steel and aluminum are not susceptible to general corrosion.

The staff recognizes that in a sheltered environment, deliquescence of airborne salts below the dew point also could generate an aqueous electrolyte initiating general corrosion. These salts may be chloride-rich and originate from marine environments, deicing salts, and condensed water from cooling towers, as well as a range of other nonchloride-rich species originating from industrial, agricultural, and commercial activities. Studies have shown that $MgCl_2$, a component

of sea salt with a low deliquescence relative humidity, would deliquesce below 52°C [126°F] under realistic absolute humidity values in nature (He et al., 2014). The heat generated by the radioactive decay of spent fuel decreases over time. Time-temperature profiles calculated for the stainless steel canister shell suggest that, while initial temperatures are high, the threshold temperature for deliquescence of some salts on the external surface of the shell could be reached during the 60-year timeframe (EPRI, 2006; Meyer et al., 2013). Because steel subcomponents exposed to sheltered environments are usually located farther away from the fuel compared to the stainless steel canister shell, they are expected to reach these threshold temperatures for deliquescence at an earlier time. As such, the potential for general corrosion of steel subcomponents exposed to a sheltered environment is present.

Because aqueous electrolytes initiating general corrosion of steels exposed to outdoor and sheltered environments are potentially present, and corrosion rates may be sufficient to affect component intended functions, the staff considers that general corrosion to be credible, and therefore, aging management is required during the 60-year timeframe. The staff finds that the applicant's conclusion is consistent with the staff's assessment of general corrosion.

Additional considerations for loss of material due to corrosion of the seismic ties

The staff requested the applicant to justify why aging management for the seismic ties for the HSM-HS was not addressed. The applicant responded that the ties were addressed in the renewal application. Furthermore, the applicant stated that the shear-tension interaction ration of 0.96, cited by the staff in the request, was based on a 10% reduction in the tensile stress area of 2.5 $in^2$ for a 2-inch diameter bolt. In other words, the applicant used a tensile stress area of 2.25 $in^2$ (2.5 $in^2$ x 0.90) instead of 2.5 $in^2$ in their calculations that produced a shear-tension interaction of 0.96. Based on this 10% reduction in bolt area, the bolt can lose 0.046 inches in thickness around the outside (0.046 in of radius), and still maintain the shear-tension interaction of 0.96. The applicant also stated that this calculated loss of material is greater that the estimated corrosion penetration for ASTM A588 material over a 60 year period. The applicant further stated that the bolts are zinc coated by electroplating. Therefore, loss of material due to corrosion is not a concern, and the seismic ties have sufficient margin to shear and tensile stress.

The staff reviewed the applicant's response and calculations and concludes that, because a loss of material due to corrosion was factored into the calculation for the structural capacity of the seismic tie, and because the loss of material considered is less than that estimated for the tie material, that loss of material due to corrosion is not a credible aging mechanism for the seismic ties.

Loss of Material Due to Crevice Corrosion – Aluminum, Carbon Steel, and Stainless Steel

The applicant stated that moist air in the absence of condensation is potentially aggressive for crevice corrosion (e.g., coastal locations, near salted roads and in the path of cooling tower drift). The applicant further stated that since the DSC decay heat will decline during the period of extended operation, the presence of moist air cannot be ruled out. The applicant identified that carbon steel, low alloy steel, stainless steel, and aluminum are all susceptible to crevice corrosion. Therefore, the applicant concluded that loss of material due to crevice corrosion is an aging effect requiring management for these HSM metals.

As discussed in the general corrosion section on page 3-44 the staff considers credible the potential to form aqueous electrolytes on surfaces exposed to outdoor and sheltered

environments, either via direct exposure to precipitation or through deliquescence of deposited salts. These electrolytes, demineralized water, and groundwater or soil could be conducive to pitting and crevice corrosion of steel. For steel embedded in concrete, as 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.

The staff notes that localized corrosion of steels is attributed to the presence of macro-galvanic cells, where local differences in electrochemical potential are created by conditions such as chemical composition differences within the steel matrix, discontinuous surface films (e.g., mill scale), and differences in oxygen supply (Revie, 2000). Because steel subcomponents exposed to outdoor and sheltered environments are likely to come into contact with aqueous electrolytes, and the localized corrosion in these environments is possible, the staff considers that loss of material due to pitting and crevice corrosion is considered to be credible. Therefore, aging management is required during the 60-year timeframe. The staff finds that the applicant's conclusion is consistent with the staff's assessment of crevice corrosion.

Loss of Material Due to Pitting Corrosion – Aluminum, Carbon Steel, and Stainless Steel

The applicant stated that areas in which aggressive species can concentrate, that is, locations of frequent or prolonged wetting or of alternate wetting and drying, are particularly susceptible to pitting. The applicant clarified that most pitting is the result of halide contamination, with chlorides, bromides, and hypochlorites being prevalent. Therefore, the applicant concluded that loss of material due to pitting corrosion is an aging effect requiring management for all HSM metals.

In the previous discussions for general corrosion and crevice corrosion, the staff has provided a basis for concluding that electrolytes may be present in aluminum, carbon steel, and stainless steel subcomponents of the HSM. Because steel subcomponents exposed to outdoor and sheltered environments are likely to come into contact with aqueous electrolytes, and the localized corrosion in these environments is possible, the staff considers that loss of material due to pitting and crevice corrosion is considered to be credible. Therefore, aging management is required during the 60-year timeframe. The staff finds that the applicant's conclusion is consistent with the staff's assessment of pitting corrosion,

Loss of Material Due to Galvanic Corrosion

The applicant stated that dissimilar metals are not used in the HSM design, therefore HSM subcomponents are not vulnerable to galvanic corrosion. The applicant clarified that dissimilar materials are in contact in the HSM at DSC support rail. In addition, the graphite lubricant used on the DSC support structure rail faces is noble relative to the rail face and the canister shell, which could induce galvanic corrosion of the DSC shell. The applicant clarified that there is no reported operating experience of stainless steel degradation where a graphite lubricant film causes loss of material due to galvanic corrosion of the DSC shell. However, the applicant recognized the potential for these aging effects and determined them to require management during the period of extended operation.

As previously discussed in this section, aqueous electrolytes for subcomponents exposed to outdoor and sheltered environments are expected to be present during the 60-year timeframe. 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 to be credible in dissimilar metal couples. Therefore, aging management is required

during the 60-year timeframe. The staff finds that the applicant's conclusion is consistent with the staff's assessment of galvanic corrosion.

Cracking Due to Stress Corrosion Cracking – Stainless Steel

The applicant stated that austenitic stainless steels and some aluminum alloys are susceptible to stress corrosion cracking (SCC). The applicant clarified that it had not identified any industry operating experience that referenced SCC in the HSM Model 152's stainless steel DSC support structure, or SCC in HSMs with stainless and aluminum steel heat shields (i.e., HSM Model 152 and HSM-H). The applicant stated that residual stresses at the welds are likely to be sufficient to initiate SCC at storage sites with sufficient chloride aerosols. There are no weld joints in the heat shields. Therefore, the applicant concluded that SCC at the welds and heat affected zones of stainless steel DSC support structures, and at the welds of the rail face to carbon steel support structures is subject to aging management.

The staff agrees that SCC of the stainless steel support structure is credible, as residual stresses may be sufficient for SCC initiation. The staff finds that the applicant's conclusion is consistent with the staff's assessment of SCC of stainless steel.

Cracking Due to Stress Corrosion Cracking – Bolting

The applicant stated that bolting fabricated from high-strength (measured yield strength, Sy, ≥ 150 ksi) low-alloy steel is susceptible to stress corrosion cracking as discussed in NUREG-1801, Revision 2 (NRC, 2010d). The applicant referenced the HSM installation specifications to define expected tensile stresses in these subcomponents. The applicant concluded that these HSM structural component anchorages do not have sufficiently high tensile stress to initiate SCC; therefore cracking of high strength bolts due to SCC is not an aging effect requiring management for HSM subcomponents.

The staff agrees that SCC of the high strength bolting is not credible, as residual stresses are expected to be insufficient for SCC initiation. The staff finds that the applicant's conclusion is consistent with the staff's assessment of SCC of high strength bolting.

Cracking Due to Thermal Fatigue – Aluminum, Carbon Steel, and Stainless Steel

The applicant stated that the only source of thermal fatigue is environmental temperature fluctuation. For HSM steel subcomponents located inside the HSM, i.e., in a sheltered environment, the applicant stated that the thermal fluctuations due to external ambient temperature fluctuations are significantly dampened by the HSM walls and roof and the DSC decay heat. In addition, the applicant provided a bounding thermal fatigue evaluation performed by considering the maximum bending moment range caused by temperature fluctuations in the thermal analyses of the HSM. Based on this analysis, the applicant concluded that thermal cycling fatigue due to fluctuations in the ambient conditions is not an aging effect requiring management for HSM subcomponents. The staff reviewed the analysis in the renewal application (see Section 3.4.3 of this SER), and finds the applicant's conclusion regarding thermal fatigue to be adequate.

Cracking Due to Thermal Cycling Fatigue of the DSC Support Structure

The applicant stated in the renewal application that the source of thermal fatigue for the DSC support structure is due to daily environmental temperature fluctuations. Because the DSC

steel support structure is located inside the HSM, the thermal fluctuations due to external ambient temperature fluctuations are significantly damped due to the HSM enclosure walls and roof as well as the decay heat from the DSC. The applicant concluded that thermal cycling fatigue due to fluctuations in the ambient conditions is not considered an aging effect requiring management for the DSC steel support structure. Since the applicant has proposed an aging management program, which includes inspections for loss of material of the DSC support structure, the staff concludes that any potential aging effects due to thermal cycling fatigue of the DSC support structure will be adequately managed.

The staff reviewed the design bases in the UFSAR, applicable industry-wide operating experience (Section 3.2 of the renewal application) and guidance provided in reports, and consensus codes and standards [ACI 349.3R-02 (ACI, 2002), ACI 201.1R-08 (ACI, 2008a) and ASCE 11-99 (ASCE, 2000), NUREG-6927 (Naus, 2007)]. The staff also used guidance in NUREG-1927, Revision 1, which references technical bases for credible age-related degradation of reinforced concrete structures. Based on its review, the staff finds the applicant's identification of aging mechanisms and effects for the HSM acceptable.

Change in Material Properties Due to Elevated Temperature – Aluminum, Carbon, Steel, and Stainless Steel

The applicant reviewed the maximum temperature of the HSM steel subcomponents at the beginning of storage for a bounding HSM model and bounding decay-heat load during off-normal conditions of storage. The applicant determined that this temperature is well within allowed temperature limits for the structural metal components of the HSMs, per the ASME Code.

The applicant provided technical bases that carbon steels and aluminum alloys are not susceptible to embrittlement due to thermal aging, and that stainless steel subcomponents remained below temperatures where embrittlement of stainless steel welds has been observed. Therefore, the applicant concluded that change in material properties due to elevated temperature is not an aging effect requiring management for HSM metal subcomponents.

The staff recognizes that undesired material property changes due to tempering of hardened steels could occur at temperatures greater than 200°C [392°F]. The temperatures of steel subcomponents exposed to sheltered air, outdoor air, demineralized water, groundwater or soil, and embedded environments are bounded by the stainless steel canister shell temperature, because these subcomponents are located farther away from the fuel. Time-temperature profiles calculated for the stainless steel canister shell estimate that the peak temperature is below 200°C [392°F] (EPRI, 2006; Meyer et al., 2013). Because the peak temperatures for steel subcomponents exposed to sheltered air, outdoor air, demineralized water, and embedded environments are below the temperature required to cause reductions in toughness, the staff considers that thermal aging is not considered to be credible for these subcomponents, and therefore, aging management is not required during the 60-year timeframe. The staff finds that the applicant's conclusion is consistent with the staff's assessment of thermal aging of metallic subcomponents in the HSM.

Change in Materials Properties Due to Irradiation Embrittlement - Aluminum, Carbon Steel, and Stainless Steel

The applicant stated that high neutron radiation can cause loss of fracture toughness in steel. Per the calculations in the renewal application, the applicant concluded that the neutron

radiation seen by the HSM steel subcomponents is orders of magnitude lower than that required to produce any effect, so neutron radiation is not a credible aging effect. The applicant further stated that gamma radiation does not have any significant impact on the properties of steel. The staff reviewed the technical basis and finds the applicant's assumptions and conclusion to be adequate since the expected neutron radiation during the period of extended operation is insufficient to cause embrittlement.

#### 3.3.3.3 Evaluation of the Proposed Aging Management Activities

The applicant proposed an aging management program, "HSM AMP for External and Internal Surfaces," to manage the identified aging effects or mechanisms for the HSM. The applicant further proposed a separate aging management program, "HSM Inlets and Outlets Ventilation Aging Management Program," to prevent heat convection capacity due to HSM inlet and outlet vents blockage. The latter AMP formalizes the requirements of CoC technical specifications, which require the licensee to monitor the HSM to identify and correct blockage of the air inlet and outlet vents that would affect the convective removal of heat from the DSC. The staff reviewed the renewal application and references therein, including design basis information in the UFSAR and relevant operating experience. The staff concludes that an AMP is acceptable for ensuring the identified aging effects will not lead to a loss of intended function. The staff review of these AMPs is located in Section 3.6 of this SER.

### 3.3.4 Storage Pad (Basemat)

The applicant stated that the HSM is installed on a load-bearing foundation, which consists of a reinforced concrete pad (also referred to as basemat) on a subgrade suitable to support the loads. There are no structural connections or means to transfer shear between the HSM base unit module and the concrete basemat. The approach slab is a reinforced concrete slab that provides access and support to the DSC transfer system. The concrete basemat and the approach slab are classified as not important-to-safety as they are not relied upon to provide safety functions. However, failure of the basemat may affect retrievability of the DSC from the HSM, therefore the applicant determined it to be within the scope of renewal. More specifically, both above-grade and inaccessible below-grade areas of storage pad are within the scope of renewal review.

The applicant clarified that the general licensee must evaluate the foundation in accordance with the requirement in 10 CFR 72.212(b)(5)(ii). The approach slabs in front of the HSMs are constructed separately from the HSM concrete pad. The transfer system is designed to accommodate any credible differential settlement between the two slabs.

The staff reviewed the UFSAR, which verifies that the storage pad and approach slabs are not important-to-safety structures, and are designed, constructed, maintained, and tested as a commercial grade items. The UFSAR further states that licensees are required to perform an assessment to confirm that the license seismic criteria are met and that the HSM foundation design meets the applicable design requirements. 10 CFR 72.212(b)(5)(ii) requires that general licensees perform written evaluations that establish that storage pads and areas are designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion. Based on its review, the staff concludes that the applicant adequately described the concrete pad design bases consistent with the UFSAR. Since degradation of the storage pad may compromise retrievability of the DSC from the HSM, an aging management review is required to address potential degradation during the

period of extended operation. The staff's evaluation of the applicant's AMP for the concrete pad is included in SER sections 3.3.4.2 and 3.6.4.

#### 3.3.4.1 **Materials and Environments**

The applicant identified the materials of construction of the storage pad as (steel) reinforced concrete. The applicant stated that the exposed exterior surfaces of the pad are exposed to all weather conditions, including insolation, wind, rain, snow, and plant-specific ambient temperature, humidity, and airborne contamination. The applicant clarified that some ISFSIs may be in environments that could be classified as corrosive based on plant operating experience, which has shown exposure to aggressive species (such as saltwater atmosphere, sulfur dioxide) in industrial areas or areas near the seashore. The areas of the pad within the footprint of the installed HSM array are in a sheltered environment. These areas are protected from outdoor effects (e.g. direct sunlight, precipitation, etc.) and experience temperatures and radiation exposure levels that are bounded by those of the HSM concrete components. The applicant further clarified that below-grade portions of the pad are in an underground environment and could be exposed to a groundwater/soil environment.

The staff evaluated the UFSAR to identify the characteristics of the service environment of the concrete pads. The staff further verified the bounding maximum temperatures in the HSM concrete at the beginning of storage, per the UFSAR, and confirmed that these temperatures did not exceed those identified in NUREG-1536, Revision 1 (NRC, 2010a), which would require additional testing. The temperatures of the concrete pads are expected to be lower than the bounding values for the HSM, which surfaces will be in closer proximity to the DSC. The staff agrees with the applicant's conclusion that these temperatures are expected to decrease over the periods of extended operation. Based on its review of the design bases in the UFSAR, the staff concludes that the applicant adequately identified the materials of construction and service environment of the concrete pad (basemat).

#### 3.3.4.2 **Aging Effects/Mechanisms for the Storage Pad (Basemat) During the Period of Extended Storage**

The applicant defined aging effects and aging mechanisms on the concrete pad (basemat) that could affect the ability to retrieve the DSC from the HSM. The cited mechanisms are consistent with those identified in the staff's example aging management program for reinforced concrete structures in Appendix B of NUREG-1927, Revision 1.

The staff reviewed the design bases in the UFSAR, applicable industry-wide operating experience and guidance provided in reports, and consensus codes and standards [ACI 349.3R-02 (ACI, 2002), ACI 201.1R-08 (ACI, 2008a) and ASCE 11-99 (ASCE, 2000), NUREG-6927 (Naus, 2007)]. Based on its review, the staff finds the applicant's identification of aging mechanisms and effects for the concrete pad (basemat) acceptable.

#### 3.3.4.3 **Evaluation of the Proposed Aging Management Activities**

The applicant proposed an aging management program, "HSM AMP for External and Internal Surfaces," to manage the identified aging effects or mechanisms for the storage pad. The staff reviewed the renewal application and references therein, including design bases information in the UFSAR and relevant operating experience. The staff concludes that an AMP with inspection, monitoring, and mitigation activities is acceptable for ensuring the identified aging

effects will not lead to a loss of intended function. The staff review of this AMP is located in Section 3.6 of this SER.

### 3.3.5 Transfer Cask (TC)

The applicant summarized the design of the transfer cask in Section 1.2.2.3 of the renewal application. The applicant stated that the transfer casks are non-pressure-retaining cylindrical vessels with a bolted top cover plate and a welded bottom end assembly. The TCs are designed and fabricated in accordance with the ASME Code Section III, Division 1, Subsection NC, “Class 2 Components,” with code alternatives described in the UFSAR. The applicant also stated that the casks provide for radiological shielding and heat transfer during transfer operations, which include DSC handling and closure at the pool, DSC transfer to the HSM, and DSC insertion into the HSM. The TC also provides structural support to protect loaded DSCs during off-normal and drop accident events.

The applicant categorized the transfer casks into five alternate configurations as summarized in Table 3.3-10. These designs are described in greater detail below in Section 3.3.5.1 of this SER.

| Table 3.3-10. Transfer Cask Designs | | |
|---|---|---|
| Design | | Features |
| 1 | Standardized Cask | Base configuration. Neutron shielding in the outer annulus is provided by NS-3 solid material |
| 2 | OS197 / OS197H | Neutron shielding in the outer annulus is provided by water. The "H" configuration includes modifications to strengthen the cask for increased lift capability |
| 3 | OS197FC / OS197H-FC<br>OS197FC-B / OS197HFC-B | The "FC" configuration includes modifications for increased air circulation to address higher heat loads. The "FC-B" configurations also include modifications to accommodate the 61BTH DSC designs |
| 4 | OS197L | A reduced-weight design that requires supplemental radiation shielding to compensate for the lack of a lead shield |
| 5 | OS200 / OS200 FC | A larger diameter configuration to accommodate larger diameter DSCs. It may include an insert for use with smaller diameter DSCs. |

#### 3.3.5.1 **Material and Environments**

Materials

The applicant provided the material specifications for the transfer cask structural materials in Tables 3B-1 and 3B-2 of the renewal application.

In Section 1.2.2.3 and Appendix 1 of the renewal application, the applicant stated that, in all configurations except the OS197L type TC, the casks are constructed from three concentric cylindrical shells: a stainless steel inner liner, a structural shell made of stainless or carbon steel, and an outer stainless steel jacket. The annulus formed by the inner liner and structural shells is filled with lead to provide gamma shielding. The annulus formed by the structural shell and outer jacket is filled with BISCO NS-3 material or water for neutron shielding. The inner liner and structural shells are welded to heavy forged stainless steel ring assemblies at the top and bottom ends. A stainless steel cover plate and elastomer O-ring seals the ram access penetration in the bottom end assembly during fuel loading. After fuel loading, this cover plate is replaced with a temporary neutron/gamma shield plug during DSC transfer to the HSM.

The OS197L type TC is unique in that it uses a single thicker shell in place of the inner and structural shells of the other designs in order to achieve a lighter weight. The absence of the lead gamma shielding annulus in this design is compensated for by the use of carbon steel plate for supplemental gamma shielding to enclose the TC during handling in the fuel or reactor building (decontamination area) and while on the transfer trailer.

All TC configurations include top and bottom trunnion assemblies that are used to lift the transfer cask and support the cask on the transfer trailer. Trunnions are fabricated from carbon steel or stainless steel and are welded to the structural shell. Some trunnion designs also

incorporate NS-3 shielding. All TC configurations also include stainless steel sliding rails that guide and support the DSC in the transfer cask cavity. The rails are coated with a graphite dry film lubricant before each loading campaign to ease the DSC transfer in and out of the cask.

Although some TC components are coated, the applicant stated that no credit is taken for the coatings for the prevention of corrosion.

Environments

In Sections 1A.1.1 and 6A.7 of the renewal application, the applicant described the operating environments for the transfer cask. The applicant identified the following normal operating environments:

- Sheltered
- External
- Water (neutron shield)

Regarding the sheltered environment, the applicant stated that, when not in use, the transfer cask should be stored in a building or container that prevents direct exposure to precipitation.

When the transfer cask is in use, the external environment of the cask varies from being submerged in water, sheltered within a building, or exposed to outdoor weather conditions. During the loading of fuel into the DSC, the transfer cask is submerged in the spent fuel pool water. The annulus between the DSC and TC during fuel loading is isolated from the spent fuel pool water to prevent contamination of the DSC with the spent fuel pool water. This annulus is filled with demineralized water and sealed prior to lowering the TC into the fuel pool. After the fuel loading, the TC is raised out of the pool and its external surfaces are rinsed with demineralized water. Finally, as the DSC is transported to the HSM, the transfer cask external surfaces are directly exposed to the weather.

The applicant stated that all transfer cask surfaces that are exposed to spent fuel pool water during fuel loading are constructed of stainless steel. The TC structural shell and the top cover plate are not exposed to fuel pool water and may be constructed from carbon or stainless steel.

The water neutron shield environment is intermittently present within the outer annulus of all transfer cask configurations other than the standardized cask, which uses solid NS-3 neutron shielding material. The applicant stated that site procedures will require that demineralized water shall be used to fill the neutron shield prior to use, but the water shall be drained after use to prevent damage due to freezing and degradation due to corrosion.

The staff reviewed the design bases of the transfer casks, including the UFSAR descriptions and design drawings, and confirmed that the materials of construction and environments were appropriately identified. The staff concludes that the applicant adequately identified the materials of construction and environments for the transfer cask subcomponents that are subject to AMR.

#### 3.3.5.2 Aging Effects/Mechanisms for the Transfer Cask During the Period of Extended Storage

In a proprietary portion of the renewal application, the applicant evaluated a range of possible aging mechanisms and effects to identify those that could, if left unmanaged, result in a loss of component intended function. In Section 6A.7 of the renewal application, the applicant identified the following aging effects for the transfer cask that require aging management:

- Loss of material due to general corrosion (carbon and low-alloy steels)
- Loss of material due to pitting and crevice corrosion (carbon, low-alloy, and stainless steels)
- Loss of material due to wear (stainless steel inner liner, trunnion bearing surfaces, TC rails, and fasteners)

The staff reviewed the applicant's evaluation of aging mechanisms and effects for the transfer cask. In its review, the staff considered NRC guidance (NRC, 2016), the technical literature, and operating experience from nuclear and nonnuclear applications (NRC, 2014; 2010c; Chopra et al., 2014; Hanson et al., 2012; Sindelar et al., 2011; NWTRB, 2010). A summary of the staff's evaluation of the identified aging effects/mechanisms follows.

General Corrosion

The staff reviewed the applicant's evaluation of the potential effects of general corrosion on carbon, low-alloy, and stainless steel transfer cask components. The staff notes that general, or uniform, corrosion of carbon and low-alloy steels in moisture-bearing atmospheres is a well-known aging mechanism. The rate of material loss depends on a number of factors, including humidity, time of wetness, atmospheric contaminants and oxidizing species (Fontana, 1986). The staff also noted that stainless steels, however, form an oxidized protective film on their surfaces that effectively prevents uniform corrosion (Grubb et al., 2005). Therefore, the staff finds the applicant's evaluation of general corrosion to be acceptable because the applicant will manage this known mechanism for carbon and low-alloy steels and has appropriately excluded the consideration of general corrosion for stainless steels, given their passivity in transfer cask environments.

Pitting and Crevice Corrosion

The staff reviewed the applicant's evaluation of the potential effects of pitting and crevice corrosion on carbon, low-alloy, and stainless steel transfer cask components. The staff notes that moisture may be present on transfer cask external surfaces due to direct exposure to the outside environment as well as during periodic immersion in the spent fuel pool. The staff also notes that the outside atmosphere can transport contaminants to transfer cask surfaces, although periodic cleaning and decontamination may prevent significant contaminant buildup. Thus, the conditions necessary for pitting and crevice corrosion may be present for carbon and low-alloy steels (Revie, 2000) and stainless steels (Grubb et al., 2005). Therefore, the staff finds the applicant's evaluation of pitting and crevice corrosion to be acceptable because the applicant's management of these aging mechanisms is consistent with their known occurrence in contaminated, moisture-bearing environments.

Galvanic Corrosion

The staff reviewed the applicant's evaluation of the potential effects of galvanic corrosion. The staff notes that galvanic corrosion occurs when two dissimilar metals or conductive materials are in physical contact in the presence of a conducting solution. The less noble of the two materials oxidizes and can experience a loss of material. The staff notes that galvanic corrosion will not occur in the absence of an aqueous electrolyte to sustain the corrosion reaction (Baboian, 2003). The staff reviewed the UFSAR, including the transfer cask drawings, and verified that there are no dissimilar metal contacts in the presence of an electrolyte during normal conditions. Therefore, because there is not an aqueous electrolyte present to sustain a corrosion reaction at the locations of dissimilar metal contacts in the transfer cask designs, the

staff finds the applicant's conclusion that galvanic corrosion is not an aging effect requiring management to be acceptable.

Wear

In Section 6A.7 of the renewal application, the applicant identified the following locations on the transfer casks where materials are in sliding contact and thus may be subject to loss of material due to wear: the stainless steel inner liner, TC rails, trunnions, and fasteners.

The staff reviewed the applicant's evaluation of the potential effects of wear. The staff also reviewed the UFSAR, including the transfer cask drawings, and verified the applicant's assessment of locations where there may be sliding contact. The staff notes that such locations of sliding contact between metals are susceptible to adhesive wear (Magee, 1992). Therefore, the staff finds the applicant's evaluation of wear to be acceptable because the applicant's management of this aging mechanism is consistent with this known degradation mode in locations of sliding contact.

Stress Corrosion Cracking

The applicant did not identify stress corrosion cracking as an aging effect requiring management for the stainless steel transfer cask components. The staff notes that austenitic stainless steels may be susceptible to chloride-induced stress corrosion cracking, particularly when the material is sensitized (Grubb et al., 2005; Kain, 1990; He et al., 2014). The staff reviewed the UFSAR and confirmed that transfer casks typically are exposed to indoor environments during their storage between cask loading campaigns, and thus an aqueous electrolyte is not likely to be present on the transfer cask external surfaces for extended periods. Also, the transfer cask external surfaces are rinsed with demineralized water to decontaminate the casks as they are removed from the spent fuel pool, which would be expected to remove any chlorides or other contaminants that may be present. Finally, the staff notes that the demineralized water within the transfer cask neutron shield is expected to be free of contaminants that could support SCC, and this shield is drained after each use. Therefore, the staff finds the applicant's conclusion that SCC is not an aging effect requiring management to be acceptable because the normal conditions of storage and the periodic cleaning of external surfaces of contaminants are expected to prevent the occurrence of this aging effect. In addition, the demineralized water environment within the neutron shield jacket similarly is not expected to support SCC.

Thermal Fatigue

The applicant did not identify thermal fatigue as an aging effect requiring management for the transfer cask. The renewal application includes the applicant's time-limited aging analysis (TLAA) of the effects of cyclic loading on the transfer cask materials in accordance with the provisions of NB-3219.2 of the ASME Code. The applicant concluded that all criteria of NB-3219.2 support a determination that a detailed fatigue analysis is not required, and thus cracking due to thermal fatigue is not an aging effect requiring management for transfer cask subcomponents.

The staff's review of the of the TLAA on the thermal fatigue of transfer casks is documented in Section 3.4.2 of this safety evaluation report.

Thermal Aging

The applicant did not identify thermal aging as an aging effect requiring management for the transfer cask. The staff notes that the microstructures of most steels will change, given sufficient time at temperature, and these changes may alter the material's strength and fracture toughness.

The staff reviewed the transfer cask materials in proximity to the DSC with respect to potential thermally-induced changes in material properties. The staff notes that the transfer cask inner liner and rails are manufactured of austenitic stainless steels. Annealing to reduce the strength of austenitic stainless steels typically is performed above 1,000°C [1,832°F], and thermally-induced embrittlement of stainless steel welds takes place above approximately 300°C [572°F] in instances where welding creates excessive amounts of the ferrite phase. These temperature thresholds of known annealing and embrittlement changes to stainless steels are well above the temperatures associated with the transfer cask inner liner described in the UFSAR. Therefore, the staff finds the applicant's conclusion that thermal aging is not an aging effect requiring management to be acceptable because the temperature exposure during DCS transport is not sufficient to cause a degradation of material properties.

Irradiation Embrittlement

The applicant did not identify irradiation embrittlement as an aging effect requiring management for the transfer cask. The staff reviewed the applicant's evaluation of the potential effects of neutron irradiation on carbon, low-alloy, and stainless steels. Neutron radiation (rather than gamma radiation) has the greatest potential to cause this phenomenon. For carbon and alloy steels, the staff notes that neutron irradiation has the potential to increase the tensile and yield strength and decrease the toughness (Nikolaev et al., 2002). Neutron fluence levels greater than $10^{19}$ n/cm$^2$ [$6.45 \times 10^{19}$ n/in$^2$] have been found to be required to produce measurable degradation of mechanical properties (Nikolaev et al., 2002; Odette and Lucas, 2001). For stainless steels, the staff notes that, depending on the neutron fluence, radiation can cause changes in stainless steel mechanical properties such as loss of ductility, fracture toughness, and resistance to cracking (Was et al., 2006). Gamble (2006) found that neutron fluence levels greater than $1 \times 10^{20}$ n/cm$^2$ [$6.5 \times 10^{20}$ n/in$^2$] are required to produce measurable degradation of the mechanical properties. Caskey et al. (1990) also indicates that neutron fluence levels of up to $2 \times 10^{21}$ n/cm$^2$ [$1 \times 10^{22}$ n/in$^2$] were not found to enhance SCC susceptibility.

For dry cask storage systems, a neutron flux of $10^4$–$10^6$ n/cm$^2$-s [$6.45 \times 10^4$ – $6.45 \times 10^6$ n/in$^2$-s] is typical (Sindelar et al., 2011). At these flux levels, the accumulated neutron dose after 60 years is about $10^{13}$–$10^{15}$ n/cm$^2$ [$6.45 \times 10^{13}$–$6.45 \times 10^{15}$ n/in$^2$], which is four to six orders of magnitude below the level that would degrade the fracture resistance of carbon, low-alloy, and stainless steels. Further, transfer casks do not experience continuous radiation exposure. Rather, they are only exposed for brief periods during DSC transfer operations. Therefore, because the neutron exposure experienced by transfer casks are significantly lower than that required to alter mechanical properties, the staff finds the applicant's conclusion that irradiation embrittlement is not an aging effect requiring management to be acceptable.

Loss of Shielding

The applicant did not identify loss of shielding as an aging effect requiring management for the lead and NS-3 shielding materials. The staff reviewed the transfer cask drawings and confirmed that the lead is fully-encased in metal and thus is not exposed to water or atmospheric contaminants. The staff notes that lead is well-known to be resistant to corrosion in a variety of environments (Alhasan, 2005), and that lead is not susceptible to thermal or irradiation-induced

material property changes under the exposures in the transfer cask application. For the NS-3 material, the staff notes that the accumulated radiation dose in storage systems after 60 years ($10^{13}$–$10^{15}$ n/cm$^2$ as described in the evaluation of metal embrittlement above) is several orders of magnitude below the 1.5 X $10^{19}$ n/cm$^2$ thermal neutron limit cited in the NS-3 product specification sheet (BISCO, 1986). The staff also notes that the shielding materials are subjected to elevated temperatures and radiation only during the relatively brief periods when the DSC is being transported from the spent fuel pool to the storage pad. Thus, the time of thermal and radiation exposure is minimal compared to the continuous exposures in other NRC-approved applications (e.g., NS-3 material in the MC-10 metal storage cask (NRC, 2005a). Therefore, because the temperature and radiation exposures experienced by transfer cask shielding materials are significantly lower than those that would be expected to alter shielding properties, the staff finds the applicant's conclusion that loss of shielding is not an aging effect requiring management to be acceptable.

#### 3.3.5.3 Evaluation of the Proposed Aging Management Activities

Time-Limited Aging Analyses

The applicant provided a time-limited aging analyses for the fatigue evaluation of the transfer casks that considered both structural and thermally-induced mechanical loading. The staff reviewed the applicant's assessment that an analysis of this potential aging effect using a TLAA is appropriate. The staff notes that, according to 10 CFR 72.3, TLAAs are calculations or analyses that are contained in the design bases, involve time-limited assumptions, consider the effects of aging for SSCs important to safety, involve conclusions related to the capability of SSCs to perform their intended safety function, and were determined by the CoC holder to be relevant in making a safety determination. NUREG-1927 states that renewal applicants can disposition an identified TLAA by demonstrating that they remain valid for the period of extended operation (NRC, 2016).

The staff reviewed the UFSAR and confirmed that a transfer cask fatigue analysis was included in the original design bases. The applicant provided an analysis to show that fatigue will not impact the safety functions of the transfer cask. Therefore, the staff finds the applicant's approach to use a TLAA to address fatigue acceptable because the applicant appropriately identified this design-basis calculation and dispositioned the analysis by reexamining the effects of fatigue to account for the additional time in storage during the period of extended operation .

The staff's review of the of the TLAA on the thermal fatigue of transfer casks is documented in Section 3.4.2 of this SER.

Aging Management Program

The applicant also developed the Transfer Cask Aging Management Program (AMP) to address loss of material due to general corrosion, pitting corrosion, crevice corrosion, and wear as described above in Section 3.3.5.2 of this SER. NUREG-1927 states that the use of an aging management program is an acceptable approach to addressing aging degradation issues, and therefore, the staff find the applicant's use of an AMP to manage the effects of loss of material of transfer cask SSCs to be acceptable.

The staff's review of the Transfer Cask AMP is documented in Section 3.6.6 of this SER.

### 3.3.6 Evaluation Findings

The staff reviewed the aging management review (AMR) provided in the renewal application and supplemental documentation. The NRC staff performed its review following the guidance provided in NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance of Dry Storage of Spent Nuclear Fuel," and relevant ISGs. Based on its review, the staff finds:

F3.1 The applicant's AMR process is comprehensive in identifying the materials of construction and associated operating environmental conditions for those SSCs within the scope of renewal and has provided a summary of the information in the renewal application and UFSAR supplement.

F3.2 The applicant's review process is comprehensive in identifying all pertinent aging mechanisms and effects applicable to the SSCs within the scope of renewal and provided a summary of the information in the CoC renewal application and UFSAR supplement.

## 3.4 Time-Limited Aging Analyses Evaluation

TLAAs are calculations or analyses used to demonstrate that in-scope SSCs will maintain their intended design function throughout an explicitly stated period of extended operation (e.g., 40 years). These calculations or analyses may be used to assess fatigue life (number of cycles to predicted failure), or time-limited life (operating timeframe until expected loss of intended design function). TLAAs should account for environment effects.

Pursuant to 10 CFR 72.3, "Definitions," TLAAs must meet all six of the following criteria:

1. Involve SSCs important to safety (ITS) within the scope of the spent fuel storage certificate renewal, as delineated in Subpart L of 10 CFR Part 72, "Approval of Spent Fuel Storage Casks," respectively;

2. Consider the effects of aging;

3. Involve time-limited assumptions defined by the current operating term, for example 40 years;

4. Were determined to be relevant by the CoC holder in making a safety determination;

5. Involve conclusions or provide the basis of conclusions related to the capability of SSCs to perform their intended safety functions;

6. Are contained or incorporated by reference in the design bases.

The staff reviewed analyses provided by the applicant in support of conclusions regarding potential aging effects for SSCs and SSC subcomponents within the scope of renewal. The staff reviewed the analyses provided by the applicant to determine those meeting all six criteria per 10 CFR 72.3 for valid time-limited aging analyses. The staff also reviewed the supplemental analyses provided by the applicant in support of the proposed AMPs.

1. Fatigue Evaluation of the Dry Shielded Canisters (TLAA)
2. Fatigue Evaluation of the Transfer Casks (TLAA)
3. Horizontal Storage Module Concrete and Dry Shielded Canister Steel Support Structure Thermal Fatigue, Corrosion, and Temperature Effects Evaluation (TLAA)
4. Dry Shielded Canister Poison Plates Boron Depletion Evaluation (TLAA)
5. Evaluation of Neutron Fluence and Gamma Radiation on Storage System Structural Materials (TLAA)
6. Confinement Evaluation of 24P and 52B Non-Leaktight DSCs (TLAA)
7. Thermal Performance of Horizontal Storage Modules for the Period of Extended Operation (Supplemental Evaluation)
8. Evaluation of Additional Cladding Oxidation and Additional Hydride Formation Assuming Breach of Dry Shielded Canister Confinement Boundary (Supplemental Evaluation)
9. Evaluation of Cladding Gross Rupture during Period of Extended Operation (Supplemental Evaluation)
10. Structural Assessment of High Burnup Cladding Performance during Period of Extended Operation (TLAA)
11. Defense-in-Depth Evaluation of Dry Shielded Canister Internal Pressures Assuming High Burnup Fuel Cladding Failure during Period of Extended (Supplemental Evaluation)
12. Defense-in-Depth Structural Evaluation of Dry Shielded Canister Confinement and Retrievability Assuming High Burnup Fuel Cladding Failure during Period of Extended Operation (Supplemental Evaluation)
13. Bounding Evaluation of Dry Shielded Canister with Reduced Shell Thickness Due to Chloride-Induced Stress Corrosion Cracking under Normal and Off Normal Conditions of Storage during Renewal Period (Supplemental Evaluation)

Based on its review of the design bases documents, the staff confirmed that the applicant identified all calculations and analyses meeting all six criteria in 10 CFR 72.3 and therefore concludes that the applicant adequately identified all TLAAs.

### 3.4.1 Fatigue Evaluation of the Dry Shielded Canisters

Section 3.5.4.2 of the renewal application summarizes the results of the applicant's TLAA on fatigue evaluation of the DSCs. The applicant provided an analysis in accordance with the provisions of ASME Code, Section III, Division 1, NB-3222.4(d) of the applicable ASME Code year (ASME, 1985a, 1999, 2000a, 2006), "Rules to Determine Need for Fatigue Analysis of Integral Parts of Vessels," and indicated that fatigue effects need not be specifically evaluated provided the six conditions contained in NB-3222.4(d), if applicable, are met. These 6 conditions are based on a comparison of peak stresses with strain cycling fatigue data and include cyclic stresses generated as a result of (1) atmospheric to service pressure cycles, (2) normal service pressure fluctuation, (3) temperature difference between startup and shutdown, (4) temperature difference in normal service, (5) temperature difference between dissimilar metals and (6) mechanical loads. The applicant provided an evaluation using these six conditions to show that the ASME Code fatigue exemption requirements are satisfied for all 11 DSC designs including 24P, 24PTH, 24PT2, 24PHB, 32PT, 32PTH1, 37PTH, 52B, 61BT, 61BTH, and 69BTH. For all designs, the pressure-retaining DSC shell and cover plate components are made of stainless steel material, SA 240 Type 304 or SA-182 Type F304 (for forgings). Both of these materials have the same mechanical and structural properties. The applicant stated that the evaluation uses bounding values for the allowable stress intensity, $S_m$, for the material at service temperature, modulus of elasticity, E, and thermal expansion coefficient, α, to cover the ASME Code editions applicable to the DSCs.

The applicant stated that for the analysis of atmospheric to service pressure cycles, the only reduction in the service pressure, or initial pressure condition, occurs gradually over the life of the canister due to reduction of decay heat that results in the cooling of the DSC. The applicant stated that this is represented in the analysis as one full cycle of pressure change. Because one cycle does not exceed the number of cycles on the applicable fatigue curve corresponding to $S_a = 3S_m$ for the material at service temperature, the applicant concluded that the first condition is satisfied for all DSCs.

The applicant stated that the analysis of normal service pressure fluctuation is directly proportional to the change in temperature. The applicant considered a bounding temperature fluctuation from -40 °F to 125 °F for normal/off-normal condition changes. Significant pressure fluctuations were determined based the allowable value of Sm, the allowable stress intensity obtained from the applicable design fatigue curve for $10^6$ cycles, $S_a$, and the design pressure of the DSC, $P_d$. Because the calculated pressure fluctuations were less than the calculated minimum value for a significant pressure fluctuation, the applicant concluded that there are no significant pressure fluctuations due to normal/off-normal ambient conditions, and that the second condition is satisfied for all DSCs.

The applicant stated that the analysis of the temperature difference between any two adjacent points during startup and shutdown was evaluated for a DSC and that the normal operational cycle occurs only once. For an operational cycle of the DSC, thermal gradients occur gradually as the fuel is loaded, after the DSC closure and TC closure as the spent fuel assemblies transfer their heat to the DSC. The cycle begins to reverse itself after the DSC is placed in the HSM and the DSC gradually begins to cool during the storage period. This cycle may be shortened if the DSC is required to be unloaded for any reason. Based on a conservative number of cycles, the applicant concluded that the required temperature difference is much greater than the maximum temperature of the DSC, and consequently, much greater than the difference between any two adjacent points on the DSCs. Therefore, the third condition is met for all DSCs.

The applicant stated that the analysis of temperature difference between any two adjacent points in normal service was evaluated by calculating a significant temperature difference based on the allowable stress intensity obtained from the applicable design fatigue curve. The analysis provided by the applicant considered the total specified number of significant temperature difference fluctuations, the coefficient of thermal expansion of the stainless steel material and the modulus of elasticity. The distance between adjacent points was determined for the smaller diameter and larger diameter DSCs. The applicant assumed a conservative annual number of ambient temperature cycles (-40 °F to 117 °F) that cause a measurable thermal gradient fluctuation. The applicant multiplied the annual cycles by 100 to determine the total number of ambient temperatures cycle that could occur in 100 years. The allowed change in temperature difference was calculated based on the allowable stress intensity, assuming the number of cycle in 100 years, the elastic modulus of the material and the thermal expansion coefficient. Since the allowed change in temperature difference between any two adjacent points on the DSC was greater than the maximum calculated temperature difference between any two adjacent points for all DSCs, the applicant concluded that the fourth condition was met for all DSCs.

The applicant stated that for the analysis of temperature difference between dissimilar metals, all integrally attached components of the DSCs are made of stainless steel material with the same mechanical and structural properties. The applicant concluded that the fifth condition is not applicable to the DSCs because the DSC confinement boundary does not include dissimilar

metals, and therefore no thermal stresses occur as a result of the differences in thermal expansion coefficients or moduli of elasticity of two different metals.

The applicant stated that the analysis of mechanical loads was based on allowable stress intensity values and that the only significant normal mechanical loads for the DSC are those associated with handling operations and the operating basis seismic event. The applicant conservatively assume a number of significant load cycles over a 100 year life. Because that specified full range of mechanical loads does not result in load stresses whose range exceeds the allowable stress intensity value obtained from the applicable design fatigue curve for the total specified number of significant load fluctuations, the applicant concluded that the sixth condition was met.

The applicant concluded that because all applicable conditions given in ASME Code NB-3222.4(d) are met for a 100-year service life for all DSCs, no additional fatigue evaluations are required.

The staff reviewed the calculations for each of the six conditions in the fatigue assessment provided by the applicant in the LRA for the NUHOMS DSCs. In addition, the staff reviewed all referenced editions of ASME Code Section III, Division 1, NB-3222.4(d). The staff notes that NB-3222.4(d)(5) specifically identifies the applicability to components fabricated from materials of differing moduli of elasticity or coefficients of thermal expansion. Because all six conditions of NB–3222.4(d) are either met or have appropriately been determined to be not applicable, the staff concludes that a detailed fatigue analyses is not required and that the design and operating conditions of the NUHOMS-24P, 24PTH, 24PT2, 24PHB, 32PT, 32PTH1, 37PTH, 52B, 61BT, 61BTH, and 69BTH DSCs will not create any potential risk of fatigue failure during 60 years of operation.

### 3.4.2 Fatigue Evaluation of the Transfer Casks

Section 3.7.5.1 of the renewal application summarizes the results of the applicant's TLAA on fatigue evaluation of the transfer casks. The applicant provided an evaluation in accordance with the provisions of ASME Code, Section III, Division 1, NC-3219.2 (ASME, 1985b, 2000b), "Rules to Determine Need for Fatigue Analysis of Integral Parts of Vessels," and stated that a detailed fatigue analysis is not required because the six conditions contained in NC-3219.2 Condition B are met. These six conditions are based on a comparison of peak stresses with strain cycling fatigue data and consider cyclic stresses generated as a result of (1) the number of full range pressure cycles, (2) the range of pressure cycles, (3) the temperature difference between any two adjacent points, (4) the range of temperature difference between any two points, (5) the range of temperature fluctuation in the vessel considering dissimilar metals, and (6) the range of mechanical loads. The applicant provided an evaluation to show that the ASME Code fatigue requirements are satisfied for the Standardized TC, all OS197 Type TC variations (which includes OS197, OS197H, OS197FC, OS197H FC, OS197FC-B, OS197HFC-B, and OS197L), and the OS200 TC designs.

In the applicant's evaluation of each of the six conditions in NC-3219.2, the Standardized TC was analyzed separately from the OS197 Type and OS200 TCs. The OS197 Type and OS200 TCs are made of the same stainless steel materials, while the Standardized TC components are made of carbon steel (including the structural shell), alloy steel, or stainless steel materials. In the fatigue evaluation, the applicant used material properties associated with the maximum temperature experienced anywhere in the TC to provide a conservative analysis.

The applicant stated that the first two conditions of NC-3219.2 address the number of full range pressure cycles and the expected design range of pressure cycles during normal service. Because the TC is not a pressure-retaining boundary, there are no stresses as a result of pressure cycles during TC operation. Therefore, the number of full range pressure cycles does not exceed the number of cycles in the applicable fatigue curve identified in ASME Section III Appendix XIV and the first condition of NC-3219.2.2 is met. The evaluation in NC-3219.2.2(b) considers: (1) design pressure, (2) stress limits based on the applicable fatigue curve of ASME Section III Appendix XIV and, (3) the design stress intensity for the material for the service temperature. Because the expected design range of pressure cycles during normal service does not exceed the specified the specified quantity in NC-3219.2.2(b) the second condition of NC-3219.2.2 is met.

The maximum temperature difference between any two adjacent points allowed by the third Code condition is based on the allowable stress intensity obtained from the applicable design fatigue curve for the specified number of startup-shutdown cycles, the coefficient of thermal expansion of the materials of construction, and the modulus of elasticity of the materials of construction. The applicant's thermal stress model demonstrated that the maximum temperature difference between any two adjacent points during normal service and during startup and shutdown is less than the code-allowable temperature difference for all TCs; therefore, the applicant concluded that the third condition is met.

The maximum change in the range of temperature difference between any two adjacent points in normal service allowed by the forth Code condition is based on the applicable design fatigue curve for the total specified number of significant temperature difference fluctuations, the coefficient of thermal expansion of the materials of construction, and the modulus of elasticity of the materials of construction. The applicant stated that a significant temperature difference fluctuation occurs only once per loading cycle, during the heating of the TCs by the spent fuel assemblies. The applicant's thermal stress model demonstrated that the maximum change in the range of temperature difference during normal service is less than the code-allowable change for all TCs; therefore, the applicant concluded that the fourth condition is met.

The fifth condition applies to components that are constructed of materials with different coefficients of thermal expansion or moduli of elasticity. In this case, the maximum total range of temperature fluctuation in the TC in normal service allowed by the Code condition is based on the applicable design fatigue curve for the total specified number of significant temperature fluctuations and the coefficients of thermal expansion and moduli of elasticity of the two materials. The applicant's thermal stress model demonstrated that there are no significant temperature fluctuations for the OS197 Type and OS200 TCs (which are constructed only of stainless steel materials). The model also demonstrated that a significant temperature fluctuation occurs only once per loading cycle for the Standardized TC, from the unloaded condition to the loaded condition. In this case, the applicant demonstrated that no locations in the Standardized TC experience a range of temperature fluctuation that exceeds that allowed by the Code condition; therefore, the applicant concluded that the fifth condition is met.

The sixth condition requires that the range of mechanical loads does not result in stress intensities that exceed those obtained from the applicable design fatigue curve for the total number of significant load fluctuations. The applicant stated that the controlling normal operating mechanical loads are on the trunnions for lifting, transfer to the HSM, and insertion into the HSM. The applicant also reviewed the structural evaluations to determine maximum stress intensities associated with normal operating loads. The applicant's analysis showed that the OS197 Type and OS200 TCs experience only two significant load fluctuations per TC use,

while the Standardized TC experiences four significant load fluctuations per TC use. In accordance with ASME Code, a mechanical load fluctuation is considered significant if the total excursion of the load stress intensity exceeds the corresponding stress value, Sa, for $10^6$ cycles on the design fatigue curve for that material. In each case, the analysis demonstrated that the load fluctuations did not result in stress intensities greater than that allowed by the Code condition; therefore, the applicant concluded that the sixth condition is met.

The applicant concluded that, because all conditions given in ASME Code NC-3219.2 Condition B were met for a 60-year service life for all TCs, no additional fatigue evaluations are required.

The staff reviewed the transfer cask fatigue analysis, in which the applicant presented detailed calculations for each of the six conditions of NC-3219.2 Condition B and verified that the transfer cask fatigue evaluation meets all six conditions in Section III NC-3219.2. The staff requested the applicant justify the assertion that the number of uses per year for a transfer cask is conservative. The applicant responded with a table of historical operation data on the transfer casks certified under CoC 1004, which indicates a maximum average annual usage is well below the number of uses assumed in the fatigue calculation. The staff determines that all conditions of ASME Code Section III, Division 1, NC-3219.2, Condition B have been met and that a detailed fatigue analysis of the transfer cask is not required.

### 3.4.3 Horizontal Storage Module Concrete and Dry Shielded Canister Steel Support Structure Thermal Fatigue, Corrosion, and Temperature Effects Evaluation

Section 3.6.4.2 of the renewal application summarizes the conclusions from the analyses to evaluate the effects of elevated temperature on the HSM concrete, fatigue of the HSM reinforced concrete and corrosion of the DSC support structure.

Effects of Elevated Temperature on HSM Concrete

NUREG-1536, Revision 1 (NRC, 2010a), provides staff guidance for acceptable temperature limits during operation of DSS concrete structures. By design, general or local concrete temperatures should be kept below 93°C [200°F] to avoid mechanical deterioration. For DSS concrete designs that satisfy additional acceptance criteria, the maximum temperature during operation can exceed 93°C [200°F] but should remain less than 149°C [300°F]. Therefore, the effects of thermal dehydration are addressed during the initial ISFSI licensing or DSS approval. The applicant demonstrated that the fuel temperature decreases over time; therefore, the staff concludes that the design temperature considerations in NUREG-1536, Revision 1 (NRC, 2010a), continue to be adequate. Thus, dehydration of the HSM concrete due to elevated temperatures is not considered to be credible, and therefore, aging management is not required during the 60-year timeframe.

Reinforced Concrete Fatigue

The applicant stated that the only source of thermal fatigue is daily and seasonal/yearly due to environmental fluctuation and that defined a maximum average daily fluctuation. Due to the high thermal mass of the HSM and low conductivity of the concrete material, the magnitude of the thermal forces that could be developed due to this temperature difference is limited and that concrete sections are assumed to crack under thermal loads, causing the reinforcing steel to carry the tension loads.

For the reinforcing steel, the applicant performed a bounding thermal fatigue evaluation by considering the maximum bending moment range caused by temperature fluctuations in the thermal analysis of the HSM. By extrapolating the curves from Figure 6 of ACI 215R-74 (ACI, 1997), the applicant computed a cumulative fatigue usage factor that is below the ACI limit of 1.0.

The staff reviewed the fatigue usage calculation and applicable portions of ACI 215R-74 (ACI, 1997) and concludes that, because the fatigue usage factor is below 1.0, thermal fatigue has a negligible effect on the HSM reinforced concrete and is not an aging effect that requires management for the HSM.

The staff notes that the thermal fatigue evaluation of the HSM reinforced concrete was not addressed in any of the UFSAR appendices. Because this mechanism is not contained or incorporated by reference in the approved design bases, pursuant to 10 CFR 70.3, this is not a TLAA. However, the staff reviewed the analysis in support for excluding the effects of this aging mechanism from being managed under an AMP. Additionally, the staff notes that the tensile strength of concrete is approximately one-tenth that of its compressive strength. For that reason, concrete is generally not credited for having any tensile strength and is assumed to crack under a tensile load, thereby causing the reinforcing steel to carry the entire tensile load.

DSC Steel Support Structure Corrosion Evaluation

The applicant stated that the DSC support structure is constructed of carbon steel for all but the HSM Model 152 which is constructed of stainless steel. Paint is applied to the steel surfaces of the DSC support structure except for the austenitic steel plates and the surface of slip critical joints. The applicant stated that these coating systems have excellent adhesion to steel and excellent resistance to alkalis and are intended to provide protection for the steel against corrosion in the harshest environments. Using an upper bound of medium corrosivity band for carbon steel and neglecting the protective coating, the applicant determined a reduction in the thickness of the steel over a period of 60 years, which the applicant concluded to be negligible.

For the HSM Model 152, the DSC support structure steel is constructed of type 304 stainless steel. The applicant stated that this material has excellent general corrosion resistance in a wide range of atmospheric environments and many corrosive media. The corrosion resistance is provided by the 18% minimum chromium content.

Regardless of the protective coatings of the carbon steel and the chromium content of the stainless steel, the applicant considered the coating integrity and loss of material due to corrosion of carbon steel and localized corrosion of the stainless steel to be aging effects that require management for all HSM models.

The staff reviewed the corrosion rate calculation for the carbon steel and computed the resulting cross sectional area and moment of inertia of the steel W section after 60 years of corrosion. The staff concludes that the effect of the loss of the material on the strength of the support structure is negligible, but that the integrity and loss of material due to corrosion of carbon steel and the localized corrosion of the stainless steel are aging effects that require management.

The staff notes that the corrosion evaluation of the DSC steel support structure was not addressed in any of the UFSAR appendices. Because this mechanism is not contained or incorporated by reference in the approved design bases, pursuant to 10 CFR 70.3, this is not a

TLAA. However, the staff reviewed the analysis in support of the proposed AMP (see Section 3.6.4 of this SER).

Steel DSC Support Structure Thermal Fatigue

The applicant stated that the source of thermal fatigue for the DSC support structure is due to daily environmental temperature fluctuations. Because the DSC steel support structure is located inside the HSM, the thermal fluctuations due to external ambient temperature fluctuations are significantly damped due to the HSM enclosure walls and roof as well as the decay heat from the DSC. The applicant concluded that thermal cycling fatigue due to fluctuations in the ambient conditions is not considered an aging effect requiring management for the DSC steel support structure.

The staff notes that thermal fatigue evaluation of the DSC support structure was not addressed in any of the UFSAR appendices. Because this mechanism is not contained or incorporated by reference in the approved design bases, pursuant to 10 CFR 70.3, this is not a TLAA. However, the staff reviewed the analysis in support for excluding the effects of this aging mechanism from being managed under an AMP.

### 3.4.4 Dry Shielded Canister Poison Plates Boron Depletion Evaluation

Section 3.5.4.5 of the renewal application summarizes the results of the applicant's TLAA for Dry Shielded Canister Poison Plates Boron Depletion. The applicant indicated that for this evaluation, an MCNP5 model of a bounding fuel assembly and basket compartment was used. The analysis used the maximum neutron source per assembly which was the contained in the 24PTH DSC. The corresponding design bases fuel in the UFSAR is the B&W 15x15 Mark B. The minimum boron content for the 24PTH DSC was used in the bounding calculation. The applicant noted that while the boron content in the 24PTH is not the lowest boron content of any DSC in the Standardized NUHOMS System, the DSCs with lower boron contents also have fuel assemblies with 10× lower neutron source strengths. Therefore, the applicant stated that the using the 24PTH was justified as bounding based on the neutron strength to boron content. The applicant stated that the results demonstrated that minimal depletion occurs with the irradiation of poison plates, and also addresses the utilization of poison rod assemblies (PRAs) in the 32PT DSC.

The applicant reported that compared to the initial amount of B-10 available, the analyses indicate that a negligible amount (not more than 0.001%) of B-10 is depleted plates throughout the period of extended operation. Therefore more than 99.999% of B-10 remains in the poison plates. The applicant stated that based on this result minimal depletion is also expected for the B-10 concentration in the 32PT PRAs. The applicant reported that because the calculation shows that boron depletion is minimal, the poison plates and PRAs will maintain sub-criticality over the desired duration of storage and boron depletion is not an aging effect requiring management.

The staff conducted an independent analysis using a bounding model with maximum source strength and slightly below minimum Boron-10 content, and confirmed that negligible amount of B-10 is depleted. Therefore, the staff concluded that the applicant's results are correct (i.e., less than 0.001% of B-10 is depleted).

### 3.4.5 Evaluation of Neutron Fluence and Gamma Radiation on Storage System Structural Materials

Sections 3.5.4.2 and 3.5.5.1 of the renewal application summarizes the applicant's TLAA on the Evaluation of Neutron Fluence and Gamma Radiation on Storage System Structural Materials. The applicant stated that irradiation embrittlement can lead to a decrease in fracture toughness of steel materials and cited literature indicating that the effects on the mechanical properties of steel are discernable at fluence levels above $1x10^{18}$ neutrons/$cm^2$ (Naus, 2007; EPRI 2007). The applicant also reported that irradiation in the form of neutrons or gamma rays can affect the concrete and reinforcing steel properties. Based on the experimental data presented in ACI SP 55-10 (Hilsdorf et al., 1978) concrete experiences a reduction in compressive and tensile strength at neutron fluence exposure levels greater than $10^{19}$ neutrons/$cm^2$. The applicant indicated that a threshold value for gamma radiation levels at which a reduction in compressive and tensile strength was observed is on the order of $10^{10}$ rads and cited information showing that for reinforcing steel, a threshold level of neutron fluence of $1\times10^{18}$ neutrons/$cm^2$ has been cited as criteria for alteration of reinforcing steel mechanical properties (Naus, 2007).

The applicant stated that aluminum alloys are used in internal components of certain DSCs. These components are typically under compression. The applicant stated that aluminum alloys have minimum neutron absorption rate and their low density results in minimum gamma heating and thus, they are capable of withstanding very high neutron exposures in plant reactor environments (e.g., spent fuel storage racks). Based on this the low neutron absorption rate, the applicant stated that the threshold levels for aluminum alloys are expected to be higher than those for steel.

The applicant reported that bounding Monte Carlo N-Particle (MCNP) (ORNL, 2006) models were used to envelop all possible geometry configurations of a DSC stored inside the HSM internal cavity. The two MCNP models used incorporate the 32PTH1 DSC and the 69BTH DSC, storing 32 PWR and 69 BWR spent fuel assemblies (SNF assemblies), respectively. Each DSC MCNP model incorporates the DSC shell assembly and the internal basket assembly and is developed based on the shielding analyses models documented in the UFSAR. The applicant reported that the BWR source is based on General Electric (GE) 7×7 fuel with 0.198 MTU and the PWR source is based on Babcock and Wilcox (B&W) 15×15 fuel with 0.490 MTU.

The applicant indicated that because the 37PTH and 69BTH DSCs hold the most FAs, these two DSCs present the bounding total source. However, the aluminum rail in 37PTH DSC reduces gamma exposure in the HSM compared to the 32PTH1, and therefore in the applicant's evaluation, the 32PTH1 is modeled for the PWR case. The applicant indicated that the calculated neutron and gamma exposures for the 32PTH1 are scaled by the 37/32 to account for the 37PTH.

The applicant stated that the neutron fluence levels are calculated at the basket center compartments and that the DSC shell and neutron fluence and gamma exposure levels are calculated at the concrete surfaces of a simplified HSM model and correspond to the concrete inside the HSM cavity. The simplified HSM model maximizes the neutron fluence and gamma heating levels on the HSM; therefore, the staff considers the results bounding to all HSM designs in the application. The applicant indicated that the neutron fluence in the DSC top and bottom shield plug is less than in the DSC shell and basket compartments. The applicant indicated that the maximum gamma exposure in the front and back ends of the HSM is significantly less than the maximum gamma heating value at the side due to the DSC shielding plugs.

The applicant reported bounding neutron fluences at the stainless steel in the center basket compartment and the DSC shell using MCNP5 computer code.

The applicant further reported maximum neutron fluences in the HSM concrete for bounding design-bases loadings of PWR and BWR fuel and concluded that neutron fluence is less than level of embrittlement of stainless steel ($1\times10^{18}$ n/cm$^2$).

The staff independently verified the applicant's conclusion by calculating neutron fluence source term using 32PTH1 DSC and the 69BTH DSC for PWR and BWR respectively as bounding conditions in the MCNP models and concluded the neutron fluence is always less than $1\times10^{18}$ n/cm$^2$. Therefore, staff concluded that the neutron fluence never reaches the level of embrittlement of stainless steel, reported to be $1\times10^{18}$ n/cm$^2$ (Naus, 2007; EPRI 2007).

The staff reviewed the applicant's TLAA on the assessment of the effects of neutron fluence and gamma radiation. Because the calculated maximum integrated neutron fluence values are well below the level of concern for embrittlement of stainless steel, reported to be $1 \times 10^{18}$ n/cm$^2$ (EPRI, 2007), the applicant concluded that change in material properties due to irradiation embrittlement is not an aging effect requiring management for DSC subcomponents. The applicant stated that steels are not affected by gamma radiation. In addition the applicant indicated that aluminum alloys have a minimum neutron absorption rate and minimum gamma heating. The staff relied on extensive studies on the effects of irradiation on stainless steel and alloy steel materials. Damage to stainless steel materials occurs at fluence levels of $>10^{20}$ n/cm$^2$. For stainless steels, the effects of increasing irradiation on changes in mechanical properties are also well characterized (Gamble, 2006). For alloy steels, Nikolaev et al., (2002) and Odette and Lucas, (2001) reported that neutron fluence levels greater than $10^{19}$ n/cm$^2$ [$6.5 \times 10^{19}$ n/in$^2$] are necessary to produce measurable degradation of mechanical properties, including decreased ductility and toughness. Farrell (1995) reported the effects of neutron irradiation on comparable aluminum alloys and showed that fluences greater than $10^{20}$ n/cm$^2$ were necessary to have marked decreases in measured ductility. Therefore, based upon the calculated neutron fluence values, the staff finds the applicant's assessment that no aging management activity is required for the effect of neutron or gamma radiation of the DSC materials is acceptable.

### 3.4.6 Confinement Evaluation of 24P and 52B Non-Leaktight DSCs

Section 3.5.5.1 of the renewal application summarizes the applicant's TLAA evaluation of the dose commitments from the 24P and 52B dry shielded canisters (DSCs) during normal, off-normal, and accident conditions from the airborne release of radioactive nuclides after 20 years of storage. The applicant included the updated analysis which uses leakage rates related to the confinement leak test acceptance criterion rather than assuming an instantaneous nonmechanistic failure and follows NUREG-1536, Revision 1 (NRC, 2010a) to evaluate dispersion factors and organ doses.

The applicant stated that all DSCs under CoC 1004, other than the 24P (standard and long), the 24PT2, and 52B have been leak tested to the $10^{-7}$ ref cm$^3$/s leak-tight standard and are, therefore, not considered in this TLAA. In addition, the applicant stated that because the 24P and the 24PT2 have the same fuel qualification table, (Table 3.1-8a AREVA TN, 2014b), the analysis of the 24P applies to both models. The TLAA provided by the applicant presents the total dose commitments from a single 24P or 52B at distances of 100 m from the DSC after 20 years of storage.

The applicant stated that the original analyses (AREVA TN, 2014a) were performed using the ORIGEN2 (Croff, 1980) computer code. The applicant stated that the re-analysis of the radionuclide inventory for the 52B was based on General Electric (GE) 7x7 fuel and the re-analysis of the radionuclide inventory for the 24P was based on B&W 15x15 fuel. The applicant reported that the fraction of this inventory that is available for release follows Table 5-2 of the NUREG-1536, Revision 1 (NRC, 2010a).

The applicant stated that using the method described in ANSI N14.5 Appendix B.4 (ANSI, 1997), the Standardized NUHOMS System Technical Specification 5.2.4(c) acceptance leak test criterion of $1x10^{-4}$ ref $cm^3/s$ (AREVA TN, 2014b), was converted to leak rates of helium at the cavity pressures and temperatures of normal, off-normal, and accident conditions at the start of storage, as given in the UFSAR Tables 8.1-6, -7, - 26, and -27. The applicant clarified that the leak rate calculation does not consider the reduced internal gas temperatures and pressures that would occur after twenty years of storage. The applicant identified that meteorology dispersion credit was calculated based on the methodology described in Regulatory Guide (RG) 1.145 (NRC, 1983) and generic parameters from the NUREG-1536, Revision 1 (NRC, 2010a) were used to determine the appropriate dispersion factor.

The applicant reported the results of the calculation in the renewal application Table 3F-5 for the DSC 52B and Table 3F-6 for the DSC 24P. For both DSCs, the calculated dose for off normal conditions does not exceed the 10 CFR 72.104 dose limits, and the calculated dose for accident conditions does not exceed the 10 CFR 72.106 dose limits. The applicant concluded that compliance to 10 CFR 72.104 and 10 CFR 72.106 was demonstrated for both the 24P and 52B DSCs, at 100 m distances after 20 years of storage, including the additional 40 year renewal period, because the source decays and the internal pressure declines over time.

In addition, the applicant discussed the amount of helium that could potentially leave a DSC after 20 year and 60 year periods. The applicant stated that calculations showed, after a 20 year period, the DSC internal pressure and temperature could be 1.34 psig and 241°F. The applicant indicated a 1.569E-5 $cm^3$/sec leakage rate at these internal conditions. In addition, the applicant assumed that this leakage rate would remain constant for the 40 year renewal period. Based on these assumptions and free volume within a 24PT2S DSC (5.83E6 $cm^3$), the applicant stated that the percentage of helium loss during a 60 year operational period would be approximately 1 percent. The staff notes that the applicant has proposed a revision to the existing CoC Technical Specification 1.2.4, which identifies a more conservative leak rate relative to the aforementioned calculation.

The staff reviewed the information provided by the applicant and determined that the applicant's dose analysis was sufficient because it considered bounding fuel content and bounding canister conditions, including higher temperature and pressures that would result in bounding potential leakage rates. In addition, the staff determined that the dispersion factor used in the dose calculation was appropriate because it was based on Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plant," and because it followed the guidance described in NUREG-1536, Revision 1 (NRC, 2010a). The staff determined that the leakage rate used as part of the dose analysis was based on ANSI N14.5, "For Radioactive Materials – Leakage Tests on Packages for Shipments," and, therefore, the staff finds that the analysis performed by the applicant adequately considers the leakage rates. In regards to the DSC helium inventory discussion, staff determined that assuming a constant leakage rate between year 20 and year 60 is conservative because the temperature and pressure inside the DSC would decrease during that period and would result in a lower

leakage rate. Staff concludes that the calculated amount of helium lost is small and any decrease in thermal conductivity would be outweighed by the large decrease in the decay heat from the stored SNF assemblies during the extended storage period.

### 3.4.7 Thermal Performance of Horizontal Storage Modules for the Period of Extended Operation

Section 3.6.5.1 of the renewal application summarizes the applicant's evaluation of the confinement weld temperatures on the outer surface of the NUHOMS® dry shielded canisters (DSCs) and the concrete temperatures over the extended period of storage as input to aging management reviews (AMRs) and time limited aging analyses (TLAAs) for the DSC and the horizontal storage module (HSM). The applicant stated that the DSCs subject to this evaluation are stored in one of the three storage module types: (1) Standardized HSM (HSM Models 80 or 102), (2) HSM-H (HSM Model 202 and HSM-HS), or (3) HSM Model 152. The applicant reported that studies of chloride induced stress corrosion cracking (CISCC) show that stainless steel welds are susceptible to CISSC when the weld temperature is low enough for deliquescence to occur.

The applicant's evaluation included the calculation of confinement weld temperatures as a function of heat loads at steady state condition and average annual temperature of 70 °F [21°C]. The analysis considered heat loads in the range of 2 kW to 22 kW for DSCs loaded in the Standardized HSM (HSM Model 80 or 102) or HSM Model 152 and in the range of 2 kW to 32 kW for DSCs loaded in the HSM-H. The applicant used half symmetric, three-dimensional CFD models generated in ANSYS FLUENT (ANSYS, 2013) to simulate the air flow and heat transfer within the each HSM type. The DSC confinement weld temperature is reported as a function of the heat load and storage time.

The applicant reported maximum DSC shell temperature and the minimum confinement weld temperatures in the renewal application Table 3G-1 for various HSM types. The minimum and maximum DSC shell temperatures as a function of time for various initial heat loads are included in the renewal application Figures 3G-5 and 3G-6. The applicant indicated that the lowest DSC confinement weld temperature is located close to the DSC top cover plates facing the HSM back wall. The applicant stated that typical temperature distributions of the HSM and DSC are presented in Figure 3G-4. The applicant reported minimum and maximum HSM concrete temperatures as a function of time for various initial heat loads and HSM models in the renewal application Figure 3G-8.

The staff reviewed the information provided by the applicant including the calculated temperatures for the DSC and HSM as a function of time for various initial heat loads and HSM models. It is noted that the applicant used the calculations in the renewal application as inputs to the DSC CISCC model and the HSM TLAA. As such, the TLAA for the Horizontal Storage Module Concrete and Dry Shielded Canister Steel Support Structure Thermal Fatigue, Corrosion, and Temperature Effects Evaluation considers the input from the evaluation of the confinement weld temperatures on the outer surface of the NUHOMS® DSCs and the concrete temperatures documented in the renewal application. Likewise the estimated crack growth rates for CISCC also consider the input from the evaluation of the confinement weld temperatures on the outer surface of the NUHOMS® DSCs and the concrete temperatures documented in the renewal application.

The staff reviewed the applicant's thermal analysis model description and boundary conditions, including the use of a half symmetric, three-dimensional ANSYS FLUENT model; the thermal

properties of the components' materials used in the models were the same as previous analyses by the applicant. Staff notes that the models, including simulating the decay heat as a volumetric heat generation of the homogenized basket region, are a reasonable representation to calculate HSM and outer canister temperatures for the renewal analysis because the purpose of the results is to determine a broad range of canister surface and HSM temperatures to confirm that DSCs are at deliquescence temperatures at the beginning of storage. The use of the model by the applicant showed that initiation of CISCC cannot be excluded during the renewal periods and thus, must be considered as an aging mechanism. The staff recognizes that the model should not be considered for other licensing actions where, for example, more detailed modeling becomes necessary to determine peak cladding temperatures.

### 3.4.8 Evaluation of Additional Cladding Oxidation and Additional Hydride Formation Assuming Breach of Dry Shielded Canister Confinement Boundary

Section 3.8.5.2 of the renewal application summarizes the applicant's supplemental evaluation for additional cladding oxidation and concurrent hydride precipitation on the cladding integrity when the DSC confinement is hypothetically breached after both 40 and 60 years of storage. The applicant indicated that the cladding materials evaluated in this evaluation include Zircaloy-2, Zircaloy-4, ZIRLO™, and M5®.

The applicant stated that the additional oxide thickness formed on the cladding surface when exposed to the ambient air and corresponding clad thinning was calculated as a function of storage time using a temperature-dependent oxidation model. The applicant compared the calculated additional oxidation with the cladding oxide thickness assumed for the structural analysis in the UFSAR. The applicant stated that the effects of alloy composition and fuel burnup on the cladding oxidation were also evaluated. In addition, the applicant estimated the effects of additional amount of hydrogen content and corresponding hydride formation by assuming absorption of hydrogen released as a result of cladding oxidation in humid air.

Calculations provided by the applicant assumed a constant cladding temperature based on the maximum temperature after storage for 40 years. Other assumptions by the applicant involve the composition of the environment inside the DSC after a breach and conditions for cladding oxidation and hydrogen uptake. The applicant assumed an initial oxide thickness of 120 μm.

The calculations by the applicant show that the additional thinning is small relative to the initial oxide thickness. The additional increase in oxide thickness resulted in an additional decrease in cladding thickness of much less than 1%. Based on the limited additional cladding oxidation, the applicant reported that the amount of additional hydride formation due to additional hydrogen incorporation will be insignificant for any impact on cladding degradation.

The staff reviewed the methodology, assumptions and conclusions of this evaluation and finds them acceptable. The staff notes, however, that the evaluation discussed above was not addressed in any of the UFSAR appendices. Because the effects of a canister breach on cladding oxidation and hydride content are not contained or incorporated by reference in the approved design bases, pursuant to 10 CFR 70.3, this is not a TLAA. However, the staff reviewed the analysis in support of the proposed AMP (see Section 3.6.1 of this SER).

### 3.4.9 Evaluation of Cladding Gross Rupture during Period of Extended Operation

Section 3.8.5.2 of the renewal application summarizes the applicant's supplemental evaluation for potential gross cladding ruptures assuming breach of the DSC and loss of the inert environment. The applicant evaluated the time necessary to cause a gross cladding failure assuming small cracks or pinholes in the cladding are present. These cladding defects are assumed to allow the fuel to be oxidized from $UO_2$ to $U_sO_8$.

The applicant conducted the evaluation with the DSC with the highest fuel cladding temperature after the initial storage period (20 years) assuming fuel with the highest initial enrichment and burnup. The applicant calculated the maximum fuel cladding temperature for times ranging from 20 to 60 years assuming air ingression into the DSC cavity. Based on this input, the applicant then calculated the time to propagate a cladding defect due to fuel pellet expansion as a result of fuel pellet oxidation. With time, the temperature of the fuel decreases and the incubation time for the propagation of cladding defects increases. Because the incubation time is strongly dependent on temperature, the applicant concluded that gross rupture of the cladding was an unlikely event even with a breach of the DSC after 20 years in service with high burnup fuel.

The staff notes that the evaluation discussed above was not addressed in any of the UFSAR appendices. Because the effects of a canister breach on fuel pellet oxidation are not contained or incorporated by reference in the approved design bases, pursuant to 10 CFR 70.3, this is not a TLAA. However, the staff reviewed the analysis in support of the proposed AMP (see Section 3.6.1 of this SER).

### 3.4.10 Structural Assessment of High Burnup Cladding Performance during Period of Extended Operation

Section 3.8.5.2 of the renewal application summarizes the applicant's TLAA that evaluates the structural performance of high burnup fuel cladding after 20 years of storage and for the duration of the CoC renewal period. The applicant's evaluations consider both pressurized water reactor (PWR) and boiling water reactor (BWR) fuel assemblies, which are evaluated for normal and off-normal storage loads in order to assess if the cladding structural integrity and SNF assembly retrievability requirements in 10 CFR 72.122(h)(1) and 72.122(l), respectively, are met for storage periods longer than 20 years. The applicant identified the applicable loads evaluated to include self-weight, internal pressure, and handling. Applicable handling loads for the SNF assembly are those associated with retrievability of the SNF assembly from the dry shielded canister (DSC) in case of repackaging or if the spent fuel assemblies need to be moved back to the spent fuel pool.

The applicant's analysis consists of determining: (1) fuel rod internal pressures, (2) fuel cladding yield stress as a function of temperature including reductions for uncertainties and (3) bounding axial and hoop direction stress in the fuel cladding due to normal and off-normal storage conditions. The applicant then compared the bounding calculated stresses to yield stress to reasonably assure integrity and retrievability of the fuel.

The available data used by the applicant to determine fuel rod pressures was applicable for maximum burnup of 60 GWd/MTU. The applicant extended the regression equation to the higher burnup of 62 GWd/MTU (limit for the approved design bases) and added one standard deviation to the calculated value at the higher burnup to account for the uncertainty of extrapolating the relationship based on available data. The applicant then determined the

bounding fuel assembly stresses considering the axial and hoop direction stresses in the fuel cladding. The applicant stated that the entire set of SNF assembly types licensed for storage in CoC 1004 is evaluated to develop a subset consisting of the bounding fuel assemblies for each class.

The applicant calculated stresses on the assembly during retrievability operations based on the normal handling inertial loads for the DSC, as defined in the UFSAR. The staff reviewed the stress analysis provided by the applicant and concludes that the bending stress values are conservative in that they are based on a 2g acceleration for lifting. A 2g acceleration for lifts corresponds to a dynamic lift factor (DLF) of 2.0. Most DLFs are between 1.10 and 1.15, so a DLF of 2.0 is conservative.

Because the bending stress is proportional to the load within the elastic range, the staff determines that scaling the bending stress from the 75g inertial loading from the UFSAR (see note 1 of Table 3J-2 and 3J-3 of Revision 1 of the renewal application) is acceptable to determine the stress as a result of a 2g inertial load. The staff reviewed the stress calculations based on the internal pressure (see note 2 of Table 3J-2 and note 4 of Table 3J-3 of Rev. 1 of the renewal application) and the bending stress calculations that the applicant used to determine the axial and hoop stresses.

The applicant compared the calculated stresses to the limiting yield stress value for irradiated cladding materials allowable under the approved design bases. The analysis shows that the most limiting case was identified for a PWR fuel (14x14 assembly). In all cases, the applicant reported that the expected stress values fall well below the yield strength of the cladding materials, providing a sufficient safety margin. Therefore, there will be no demand on the ductility of the cladding material.

Because the axial and hoop stresses are significantly less than the yield stress of the cladding, the staff finds that the fuel can be safely handled and retrieved after 60 years of storage. The staff further determined that the decrease in the fuel rod internal pressure that is expected as a result of the decreasing temperature has a beneficial effect on the stresses in the fuel rods, because the lower pressure will cause less stress on the material.

Consideration of Radial Hydrides

The applicant concluded that the decreasing temperature of the fuel could result in a ductile-to-brittle transition. However, any reduction in ductility due to radial hydrides is not a concern due to the low stress levels during the period of extended operation. Therefore, the applicant concluded that any possible ductile-to-brittle transition of the cladding will not affect the ability to safely retrieve the SNF assemblies after 60 years of storage. The applicant also stated that the decrease in fuel rod internal pressure that is expected as a result of the decreasing temperatures also decreases the degree of precipitation of radial hydrides during the period of extended operation.

The staff reviewed the information provided by the applicant and notes that the cladding structural analysis was made using available mechanical properties, which only account for circumferential hydrides. The staff notes that potential degradation of mechanical properties due to hydride reorientation is only expected to potentially compromise the ability to maintain the analyzed fuel configuration during pinch-type loads. These loads are only expected during fuel retrieval operations, if the design bases of the dry storage system rely on retrievabilty of the high burnup fuel on a single-assembly basis (as it is in the approved storage systems under this

CoC). Pinch-type loads are not expected to be present during normal, off-normal, and accident conditions of storage. More specifically, the tensile stress field associated with potential inertial rod bending during storage is expected to be parallel to both radial and circumferential hydrides and not expected to compromise the structural integrity of the cladding.

The staff has proposed two alternatives for demonstrating that the safety analyses pertaining to the analyzed spent fuel configuration will not be compromised by the effects of hydride reorientation. The first approach relies on the applicant performing a defense-in-depth analysis, assuming credible reconfiguration based on 1 percent fuel failure for normal conditions of storage, 10 percent failure for off-normal conditions of storage, and 100 percent or other justifiable value for accident conditions. The staff has issued a generic consequence analysis for both vertical and horizontal storage configurations in NUREG/CR-7203 (Scaglione et al., 2015), which can be used by applicants in the development of their defense-in-depth analysis. A second approach relies on the evaluation of data from a demonstration (surrogate) program consistent with the guidance in Appendix D of NUREG-1927, Revision 1 (NRC, 2016). For example, destructive examination from the DOE/EPRI cask demonstration project (EPRI, 2014) may be used as confirmation that hydride reorientation has not compromised the ability to retrieve the spent fuel on a single-assembly basis. The applicant chose to use the second approach consistent with the guidance in Appendix D of NUREG-1927, Revision 1, by providing an AMP with aging management activities to confirm that the SNF assemblies remain in the analyzed configuration. The staff's review of this AMP can be found in Section 3.6.1 of this SER.

### 3.4.11 Defense-in-Depth Evaluation of Dry Shielded Canister Internal Pressures Assuming High Burnup Fuel Cladding Failure during Period of Extended

Section 3.5.5.1 of the renewal application summarizes the applicant TLAA that evaluates the confinement boundary internal pressure for the NUHOMS® dry shielded canisters (DSCs) loaded with high burnup fuel assemblies assuming that high burnup fuel rods will rupture and release fill and fission gases into the DSC cavity after the initial licensing period. The applicant indicated that the calculated DSC internal pressures are used to demonstrate that the structural integrity of the DSC confinement boundary is maintained as a defense–in-depth evaluation. The applicant also noted that this calculation determines the bounding normal and off-normal internal pressures in the NUHOMS® DSCs for the following conditions (1) Normal condition after 20 years of storage assuming that 1% of fuel rods rupture, (2) Off-normal condition after 20 years of storage assuming that 10% of fuel rods rupture, and (3) Normal condition after 60 years of storage assuming that 1% of fuel rods rupture. The applicant identified that the bounding normal and off-normal pressures after 20 years of storage are used in a defense-in-depth evaluation to calculate the critical DSC shell crack size caused by chloride-induced stress corrosion cracking (CISCC). The calculation of the critical crack size is also a defense-in-depth evaluation.

The applicant identified that the governing DSC component to demonstrate structural integrity of the DSC confinement boundary due to internal pressure is the inner top cover plate. The various NUHOMS® DSC models that are allowed for storage of high burnup fuel were categorized into four main groups according to their inner top cover plate thickness. The applicant stated that the DSC with the highest accident pressure was selected from each of the four categories for further evaluation.

The DSC internal pressures are calculated using the ideal gas law following the methodologies described in the UFSAR. The applicant identified the following assumptions for the calculation:

1. For Appendix M analysis, all fuel rods rupture after 20 years of storage.
2. For Appendix N analysis, 1% and 10% of the fuel rods are ruptured for normal and off-normal conditions, respectively, after 20 years of storage to provide the bounding DSC internal pressures for a defense-in-depth evaluation of the critical crack size due to CISCC.
3. 1% of the fuel rods are ruptured for normal conditions after 60 years of storage to provide the bounding DSC internal pressures for a defense-in-depth evaluation of the dose rate effects.
4. For ruptured fuel rods, the release rate of fuel rod fill gas and the release rate of the fission gases is consistent with NUREG-1536, Revision 1 (NRC, 2010a).
5. The annual average ambient temperature of 70 °F is considered to evaluate the DSC cavity gas temperature for normal conditions after 20 or 60 years of dry storage. The 47 °F temperature difference between the highest off-normal ambient temperature of 117 °F and the annual average ambient temperature of 70 °F is added to the average cavity gas temperature for normal condition to determine the bounding cavity gas temperature for the off-normal condition.

The applicant stated the heat loads after 20 or 60 years of storage are calculated assuming the maximum burnup (62 GWd/MTU for 61BTH Type 1 and 32PTH1-S DSCs and 55 GWd/MTU for 32PT-S125 DSC), and the maximum initial enrichment of 5 wt. % U-235. The applicant identified that the DSC shell temperature profiles are taken from the evaluations performed for each applicable horizontal storage module (HSM) and DSC type.

In regards to the assumptions listed in items 1 through 4 above, staff concludes that the assumed percentage of fuel rod ruptures and the release rate of the fuel rod fill gas and fission gas are appropriate because the values are consistent with the guidance provided in NUREG-1536, Revision 1 (NRC, 2010a). In regards to the fifth assumption, staff notes that, for this renewal application, adding 47°F to the normal condition gas cavity temperature is a simplified assumption to calculate a cavity gas temperature at a high off-normal ambient temperature and the resulting internal DSC pressure. Staff recognizes that this methodology should not be considered for other licensing actions where, for example, more detailed modeling becomes necessary to account for temperatures at maximum ambient conditions and resulting peak cladding temperatures when determining internal structural temperatures and pressures for release calculations.

### 3.4.12 Defense-in-Depth Structural Evaluation of Dry Shielded Canister Confinement and Retrievability Assuming High Burnup Fuel Cladding Failure during Period of Extended Operation

Section 3.5.5.1 of the renewal application summarizes the applicant TLAA that evaluates the dry shielded canisters (DSCs) loaded with high burnup fuel assemblies for normal and off- normal conditions. In this calculation the applicant assumed, consistent with the guidance in NUREG-1536, that there is a 100% rupture of high burnup fuel after 20 years of storage. The applicant noted that the structural evaluations of 61BTH Type 1 and 32PT are documented in UFSAR (AREVA TN, 2014a) Sections T.3 and M.3, respectively. In the calculation included in the renewal application, the DSCs are re-evaluated for the higher pressures and lower thermal loads after 20 years with 100% rupture of the high burnup fuel. The renewal application includes a calculation of the internal pressures with reduced temperature profiles after 20 years of storage and 100% rupture of the high burnup fuel. The applicant identified that the 61BTH Type 1 and 32PT-S125 have the highest internal pressure so the evaluation is performed for these two types of DSCs.

The applicant's calculation considers stresses from internal pressure and thermal stresses. The DSCs are evaluated for internal pressure load on the confinement boundary which is defined by the DSC shell, the inner bottom cover plate, the inner top cover plate, the siphon/vent block, the port covers, and the associated welds. The stress intensity plot for the 61BTH Type 1 DSC top end is shown in Figure 3M-1, and for the bottom end in Figure 3M-3. The stress intensity plot for the 32PT DSC top end is shown in Figure 3M-5, and for the bottom end in Figure 3M-7.

The applicant evaluated the thermal stresses resulting from thermal expansion and gradients between the various DSC components. The applicant noted that these stresses are classified as secondary (Q) stresses that need be evaluated only for Service Level A and B conditions. The applicant reported temperature and stress intensity distribution plots for the 61BTH Type 1 DSC top end are shown in Figure 3M-2, and for the bottom end in Figure 3M-4, and temperature and stress intensity distribution plots for the 32PT DSC top end are shown in Figure 3M-6, and for the bottom end in Figure 3M-8.

The applicant indicated that all the load combinations listed in UFSAR (AREVA TN 2014a) for the 32PT and 61BTH Type 1 DSCs are re-evaluated for higher internal pressure and reduced thermal loads after 20 years of storage. The applicant reported that the allowable stress values for the material are taken at 300 °F for the 32PT DSC and 61BTH Type 1 DSC because this is a bounding temperature for both DSCs after 20 years of storage. The applicant reported that for the confinement boundary components listed in Table 3M-2 and Table 3M-3 for the 32PT, and Table 3M-4 and Table 3M-5 for the 61BTH Type 1, the calculated stress intensity does not exceed the allowable stresses. The applicant concluded that based on the evaluation performed, the DSCs listed in CoC 1004 are structurally adequate for a hypothetical condition of 100% rupture of high burnup fuel assemblies after 20 years of storage.

The staff reviewed analysis of the stresses in the 61BTH Type 1 DSC and the 32PT DSC. The staff also reviewed the UFSAR to assess the assertion by the applicant that the stresses in the 32PT DSC (Group 2) bounded those of the 32PTH1 (Group 4) even though the internal pressure is lower. Based on the UFSAR, the stresses in the 32 PT DSC are higher than those of the 32PTH with a corresponding lower design pressure; therefore, the staff concludes that the 32PT DSC bounds that of the 32PTH. The staff reviewed the properties of the ANSYS contact elements CONTAC49 (old version) and CONTA173 (current version) and determined that they are essentially the same element and replacing the CONAC49 element with the surface-to-surface CONTA173 element does not alter the results of the model. In Tables 3M-1 and 3M-2, the highest stress ratio (actual stress/allowable stress) for the 32PT is 0.90 and occurs in the top half of the shell. In Table s 3M-3 and 3M-4, the highest stress ratio for the 61BTH is 0.95 and occurs in the outer top cover plate. Because all stress ratios in Tables 3M-1 through 3M-4 are less than 1.0, the staff concludes that the DSCs associated with the CoC 1004 are structurally adequate for a hypothetical condition of 100% rupture of high burnup fuel assemblies after 20 years of storage.

### 3.4.13 Bounding Evaluation of Dry Shielded Canister with Reduced Shell Thickness Due to Chloride-Induced Stress Corrosion Cracking under Normal and Off Normal Conditions of Storage during Renewal Period

The renewal application includes a TLAA that evaluates the minimum thickness of the dry shielded canister (DSC) shell required to demonstrate that the confinement function of the DSC is maintained and the requirement for ready retrieval of the DSC from the horizontal storage module (HSM) is met. The applicant stated that the confinement function is considered to be met if the DSC confinement boundary stresses due to normal and off-normal loads meet the ASME Code stress limits for Level A and B conditions (ASME, 2000a). The applicant indicated that the minimum thickness is governed by the crack depth postulated to occur due to chloride-induced stress corrosion cracking (CISCC).

The applicant identified 32PTH1 DSC as the bounding DSC for this evaluation based on shell thickness, bottom cover assembly thickness, transfer push and pull loads, design pressure and heat load. The evaluation was performed on the 32PTH1 DSC for normal and off-normal storage loads and load combinations (Level A and B) for retrieval of the DSC from the HSM module and the confinement boundary was evaluated based on the stresses due to maximum internal pressure and pull loads. The applicant noted that the welds of the DSC outer bottom cover plate and grapple ring were not evaluated because they were readily accessible for inspection and, if necessary, repair at the time of DSC retrieval.

The UFSAR describes the analysis, loads, and load combination results for 32PTH1 DSC. Based on the review of results for different load combinations in the UFSAR, HSM unloading combination UL-6 and HSM loading combination LD-5 were determined by the applicant to be bounding for Level A and B limits for the DSC shell. However, the applicant noted that because the purpose of this appendix is to evaluate retrieval rather than loading, load case LD-5 was ignored, and HSM unloading combination (UL-6) for the retrieval of the DSC from the HSM module was considered for the analysis.

The applicant conducted two analyses for the 32PTH1 using a finite element model (FEM) description for the DSC provided in the UFSAR. In the first analysis, the applicant reduced the DSC total thickness by half which bounds the degradation caused by CISCC cracking. Even with a 50% reduction in shell thickness, the applicant showed (Table 3N-1) that the calculated stress intensity for service level B conditions did not exceed the allowable stress intensity. The applicant provided an additional calculation to determine the minimum shell thickness necessary HSM unloading combination UL-6. In this evaluation, the applicant reported that that the service level B loading conditions could be met with a shell thickness reduced to 20% of the original thickness (Table 3N-2).

The staff reviewed the methodology used by the applicant to determine performance of the DSC with reduced shell thickness due to CISCC under normal and off normal conditions of storage during renewal period. Because the solution converges for all analyzed thicknesses, the staff determines that the ANSYS solution is acceptable. The staff notes that reducing the entire thickness of the canister shell by 50% is more conservative than introducing a 50% through wall crack in the shell that leaves half of the material remaining to resist the pressure induced load. Due to fracture mechanics, the stress at the tip of the crack will be greater than if the entire shell thickness is reduced, as the applicant assumed in their evaluation. However, because of the ductility of the stainless steel material, the higher stress at the tip of the crack will be “absorbed” by the remaining material through localized plastic deformation and strain hardening in the immediate area of the crack, and the higher stress will have little effect on the shell’s ability to

resist the load due to the internal pressure. For this reason, the staff finds that reducing the shell thickness is an acceptable method for analyzing the effects of a crack in the stainless steel shell. As a result, the staff finds that the confinement function of the DSC is maintained and the applicant has met the requirement for ready retrieval of the DSC from the horizontal storage module (HSM) for the renewal licensing period

### 3.4.14 Evaluation Findings

The staff reviewed the TLAAs provided in the renewal application. The staff performed its review following the guidance provided in NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," and relevant ISGs as identified in Table 1.6-1. The staff verified that the TLAA assumptions, calculations, and analyses were adequate and bound the environment, and aging mechanisms or aging effects for the pertinent SSCs. Based on its review, the staff finds:

F3.3 The applicant identified all pertinent aging mechanisms and effects pertinent to SSCs within the scope of renewal that involve TLAAs. The methods and values of the input parameters for the applicant's TLAAs are adequate. Therefore, the applicant's TLAAs provide reasonable assurance that the SSCs will maintain their intended function(s) for the period of extended operation, require no further aging management activities, and meet the requirements in 10 CFR 72.240(c)(2).

## 3.5 Lead Canister Inspection

The applicant stated that a lead canister inspection (referred to as "pre-application inspection" in NUREG-1927, Revision 1) was not performed prior to the renewal submittal for the NUHOMS® DSRS loaded vendor Certificate of Compliance (CoC) 1004. The applicant indicated that as a CoC holder, AREVA Inc. does not have direct authority over general licensee storage systems. The applicant did consider lead canister inspection reports and baseline inspections from similar NUHOMS®-based site-specific independent spent fuel storage installation (ISFSI) renewal applications including the Calvert Cliffs Nuclear Power Plant ISFSI (Calvert Cliffs, 2012), Oconee Nuclear Station ISFSI (Duke Energy Carolinas, LLC, 2009: U.S. NRC, 2009), and the H.B. Robinson Steam Electric Plant ISFSI (H.B. Robinson Steam Electric Plant, 1994; 2000; U.S. NRC, 2005b). Information from these NUHOMS®-based site-specific ISFSI lead canister and baseline inspection reports were evaluated as part of the aging management review (AMR) process to confirm applicable actual or potential aging effects and associated aging mechanisms specific to the NUHOMS® storage system components. The applicant indicated that the review of previous NUHOMS®-based lead canister and baseline inspections contributed to the identification of applicable aging effects in the AMR and determination of aging effects that should be managed for the period of extended storage. In addition, the review aided in the development of aging management program (AMP) inspection scope, evaluation of parameters, detection of aging effects, monitoring and trending, and acceptance criteria elements of an AMP.

The applicant stated that the CoC 1004 renewal application proposes license conditions that require general licensees to perform dry shielded canister (DSC) and horizontal storage module (HSM) AMPs. These AMPs include a requirement for baseline inspections (a lead canister inspection at the licensee's ISFSI) prior to entering the period of extended storage or as prescribed in the baseline inspection implementation schedule detailed in the renewal application Appendices 6A.3 through 6A.5. The applicant stated that the combination of incorporating operating experience (OE) learned from previous NUHOMS®-based inspections

and including a requirement for a baseline inspection at general licensee's ISFSI meet the intent of NUREG-1927 Appendix E guidance and is in compliance with 10 CFR 72.240(c)(3) and 10 CFR 72.240(d).

Calvert Cliffs Nuclear Power Plant ISFSI, Material License No. SNM-2505

The Calvert Cliffs ISFSI is located on the east side of the Chesapeake Bay and is considered to be in a coastal brackish environment. The Calvert Cliffs ISFSI was originally licensed with the NUHOMS®-24P dry storage system in November 1992. Calvert Cliffs performed an inspection of the interior of two HSMs, and the exterior surfaces of the of the DSCs on June 27th and 28th, 2012. HSM-15, loaded in November 1996, contained the lead canister DSC for the purpose of meeting the NUREG-1927 Appendix E guidance (NRC, 2011a). HSM-1, loaded in November 1993, represents one of the lowest heat load canisters presently loaded (estimated at 4.2 kW). This canister was added as part of the Electric Power Research Institute (EPRI) research efforts on evaluating stress corrosion cracking of stainless steel canisters used for dry storage. This inspection included salt concentration measurements on the upper shell of the DSC, collection of samples of the deposits on the upper shell of the DSC for offsite analysis, and surface temperature measurements via contact thermocouple for the purpose of benchmarking best-estimate thermal models.

The visual inspection was conducted in both HSM-15 and HSM-1 by remote and direct means. The remote inspection was performed by lowering a remote controlled, high definition pan-tilt-zoom (PTZ) camera system with a 100 mm head camera inserted through the rear outlet vent which allowed viewing of the majority of the DSC, its support structure, and the interior surfaces of the HSM. The direct inspection was performed through the partially open door by mounting the camera on a pole. This allowed for views of the bottom end of the DSC, the seismic restraint, HSM doorway opening, and the backside of the HSM door. Varying levels of camera magnification were utilized to highlight various areas of interest during the inspection.

Based on the visual examinations described above, it was concluded that the Calvert Cliffs baseline inspection of HSM and DSC structures were performing as expected. The inspection did not indicate any aging-related deficiencies with the DSC components. Some minor degradation on HSM external concrete surfaces was noted. There was evidence of localized water intrusion to the interior of the HSM in the vicinity of the rear outlet vents. A coating of dust and dirt was present on the floors of each HSM but no debris or standing water was observed. There was some general surface corrosion noted on the carbon steel surface and bolting hardware. Noted deficiencies have been entered into corrective action program for evaluation.

No aging-related degradation was observed for the concrete on the interior of the HSM, however, water marks on the DSC and concrete leaching where it appeared that wind-blown water had entered through the rear vent. Coating failure and corrosion on the internal HSM steel was observed and coating failure and corrosion on welds attaching one angle support for the support rails structure and one of the attachment bolts was noted. Minor corrosion spots on the canister were indicative of carbon steel contamination during fabrication or installation. The loose dry deposit found on the top of the canister was investigated for chloride content.

Oconee Nuclear Station ISFSI, Material License No. SNM-2503

The Oconee Nuclear station ISFSI is located in northwestern South Carolina on the eastern shore of Kelly Creek and Lake Keowee and is considered an inland site. The Oconee Nuclear Station ISFSI was originally licensed on January 29, 1990. The initial phase (Phase I) of

construction, which included twenty modules, was completed in May 1990. Phase II of twenty modules was completed in January 1992. A baseline civil/structural Inspection of the Oconee Nuclear Site ISFSI Phase I and II structures was conducted on December 4th and 5th, 2006. The purpose of these inspections was to evaluate the current condition of the structures as part of the site specific ISFSI License Renewal Project. The entire exterior of both structures was examined in detail. The interior of three HSMs was examined to the extent possible using remote and direct methods. One door was raised for direct inspection of the HSM opening. The conclusion was there were no indications of structural distress or degradation that would render the facility incapable of performing its intended function.

No aging-related degradation was identified for the concrete surfaces on the interior of the HSM. However, coating failure and corrosion on the internal HSM steel structure was observed. No corrosion was observed on the support brackets for the DSC support structure. Coatings had failed and there was surface corrosion on the attachment welds for the support angle supporting the front inlet plenum. Paint-flaking and corrosion were noted on the door support frame and exterior steel surfaces of some doors. The DSCs were not inspected. Based on the visual examinations described above, it was concluded that the Oconee Nuclear Site ISFSI Phase I and II structures were performing as expected. Some minor maintenance was recommended to address limited loss of coatings and corrosion, and missing alignment targets.

H.B. Robinson Steam Electric Plant ISFSI, Material License No. SNM-2502

The H.B. Robinson Steam Electric Station ISFSI is located in north central South Carolina on the south west shore of Lake Robinson and is considered to be an inland site. The H.B. Robinson Steam Electric Station ISFSI was originally licensed on August 31, 1986. Two interior remote inspections of the HSM were performed using a video camera. The first inspection was performed in 1993 as supporting documentation for a license amendment. The second inspection was performed in March 1999 as a 10-year follow-up licensing commitment. Both inspections focused on the HSM inlets and outlets to ascertain that there was no blockage internal to the HSM. No inspections of the DSC were performed.

NRC Staff Review of Lead Canister Inspection

The staff reviewed the applicant's summary of previously conducted system inspections. The staff recognizes that a CoC holder does not have the authority or access to conduct inspections of systems in service. As noted by the applicant, the renewal application proposes license conditions that require general licensees to perform dry shielded canister (DSC) and horizontal storage module (HSM) AMPs. The applicant indicated that these AMPs include a requirement for baseline inspections (a lead canister inspection at the licensee's ISFSI) prior to entering the period of extended storage or as prescribed in the baseline inspection implementation schedule detailed in the renewal application Appendices 6A.3 through 6A.5. The staff have reviewed the proposed license conditions and concludes that the baseline inspections conducted in accordance with the AMPs in the renewal application Appendices 6A.3 through 6A.5 will provide the system condition information to meet the intent of a lead system inspection as described in NUREG-1927 (NRC, 2011a). In addition, the staff finds that the requirement to perform a baseline inspection at each general licensee's site prior to entering the period of extended operation will provide significant component specific operational experience that can be utilized to inform subsequent inspections for monitoring and trending of aging effects as well as future baseline inspections at other general licensees' sites. The staff finds that the applicant's approach to require baseline inspections at each generally licensed site is adequate.

## 3.6 Aging Management Programs

Pursuant to 10 CFR 72.240(c)(3) requirements, the CoC holder must provide a description of AMPs for management of issues associated with aging that could adversely affect SSCs important to safety. The applicant provided six AMPs in the renewal application to address aging effects for the DSC, TC, HSM, storage pad and SNF assemblies.

The staff conducted the safety review for the AMPs in the application per the guidance in NUREG-1927, Revision 1 (NRC, 2016). The staff notes that Appendix B of NUREG-1927, Revision 1, provided example AMPs for certain aging effects in common SSCs important to safety. Therefore, the staff reviewed AMPs in the renewal application for consistency with Appendix B of NUREG-1927, Revision 1. If an example AMP is not provided in NUREG-1927, Revision 1, the staff has provided a more detailed discussion of the review of all 10 AMP elements per the guidance in Section 3.6.1 of NUREG-1927, Revision 1.

### 3.6.1 High Burnup Fuel Aging Management Program

The applicant credited the "High Burnup Fuel Aging Management Program" to provide the confirmatory data that high burnup fuel stored in CoC 1004 (amendments 6, 8-11, 13, 14) has reasonably maintained the analyzed configuration beyond 20 years of storage. The applicant followed the guidance in Appendix D of NUREG-1927, Revision 1 for defining an acceptable surrogate program for monitoring and assessing the condition of the stored high burnup fuel.

Appendix B (Table B-3) of NUREG-1927, Rev. 1 provides an example AMP for High Burnup Fuel Monitoring and Assessment. The staff developed this AMP per the guidance in Appendix D of NUREG-1927, Revision 1. The example AMP in NUREG-1927, Rev. 1 states that new data and research on fuel performance from both domestic and international sources that are relevant to the certified high burnup fuel in the dry storage system should be evaluated on a periodic basis and the AMP updated and revised as needed. Consistent with the review guidance, the applicant stated that the AMP will be updated as new information becomes available.

The staff reviewed the description in the 10 elements of the applicant's High Burnup Fuel AMP, and finds that the AMP provided the information expected per the example AMP in Table B-3 of NUREG-1927, Revision 1. More specifically, the applicant adequately described the design-bases high burnup fuel, the surrogate program that will be used to provide data on the applicable design-bases high burnup fuel performance, and how the parameters of the surrogate program are applicable to the design-bases high burnup fuel. The applicant adequately described the technical specifications that assure that the high burnup fuel is stored in a dry inert environment, consistent with the surrogate program. The applicant provided adequate acceptance criteria, which is justified per the design-bases and the identified aging mechanisms. The staff finds the acceptance criteria to be consistent with that specified in Table B-2 of NUREG-1927, Revision 1. The applicant adequately described corrective actions to be taken in the event that data from the surrogate program indicates that the AMP acceptance criteria has not been met. The applicant stated that corrective actions shall be implemented if data from a surrogate demonstration program or other sources of information indicate that any of the acceptance criteria in the AMP are not met. If any of the acceptance criteria are not met, the general licensee, in coordination with the CoC holder, will:

1. Assess the fuel performance for impacts on fuel and changes to the fuel configuration; and as part of the corrective actions, determine measures that should be taken to:

    a. Manage fuel performance, if any; or
    b. Manage impacts related to degraded fuel performance to ensure that all intended functions for the dry storage system are met.

    The applicant stated that NRC approval will be requested for any modifications of the design bases that are required to address any conditions outside of the approved design bases.

2. Assess the design-bases safety analyses, considering degraded fuel performance and any changes to fuel configuration, to determine the ability of the dry storage system/ISFSI to continue to perform its intended functions under normal, off-normal, and accident conditions.

The staff finds that, because each general licensee has an existing, approved Corrective Action Program, adequate assurance exists that these corrective actions will be taken in a timely manner.

The applicant concluded that the High Burnup Fuel Aging Management Program is a “learning” AMP that is designed to update/revise the approach to aging management of high burnup fuel to reflect the HDRP findings as they become available, as well as to incorporate other sources of information, including operational experience and industry research.

The applicant provided tollgates for the High Burnup Fuel AMP. The applicant defined a tollgate as a requirement included in a renewed CoC and associated UFSAR for the licensee to perform and document an assessment of the aggregate impact of age-related dry storage operating experience, research, monitoring, and inspections at specific points in time during the renewed operating period. The applicant stated that if, at the second or subsequent High Burnup Fuel AMP tollgates, the assessment confirms the ability of the high burnup SFAs to perform their intended functions for the remainder of the period of extended operation, subsequent assessments may be cancelled. In addition, the applicant described adequate corrective actions to be taken if, at any tollgate, the information available is insufficient to demonstrate the ability of the high burnup SFAs to perform their intended functions through the period to the next tollgate. The applicant further stated that the tollgate schedule may be accelerated (i.e., the next tollgate is performed earlier) whenever sufficient new information has accumulated that could warrant a change in the AMP.

The staff reviewed the applicant’s description of the tollgates for the high burnup SFAs. The staff find that the applicant has identified relevant information sources to be reviewed as part of evaluating the effectiveness and, if necessary, revising the High Burnup Fuel AMP. The applicant has stated that the information obtained in the HDRP (EPRI, 2014) or an alternative program meeting the guidance in Appendix D of NUREG-1927, Revision 1 will be used to inform the AMP. In addition, the CoC holder will update the AMP with information obtained from published research on high burnup fuel performance. The staff reviewed the applicant’s AMP tollgates and finds that the information sources referenced in the renewal application and the actions identified at each tollgate provides reasonable assurance that the AMP is adequate for proving confirmation that configuration of high burnup SFAs is maintained throughout the period of extended operation. The staff further finds that the applicant provided an adequate high burnup fuel AMP consistent with the guidance in Appendix B of NUREG-1927, Revision 1.

### 3.6.2 DSC External Surfaces Aging Management Program

The applicant has stated that the DSC External Surfaces Aging Management Program (AMP) applies to CoC 1004 Amendments 0 through 11, 13, and 14 (upon approval) and applies to all sites except those where the Effects of Chloride-Induced Stress Corrosion Cracking (CISCC) AMP applies. The applicant identified the materials and environments included in this AMP as the DSC shell assembly components constructed of stainless steel exposed to a sheltered environment inside the HSM. Aging effects requiring management identified by the applicant include (1) loss of material due to crevice and pitting corrosion for stainless steel components and (2) cracking due to stress corrosion cracking (SCC) for stainless steel components when exposed to moisture and aggressive chemicals in the environment.

The staff reviewed the narrative included in the 10 elements of the DSC External Surfaces AMP and finds that the AMP follows the guidance provided in NUREG-1927 Revision 1 Section 3.6 and the description of each AMP element is consistent with the Localized Corrosion and Stress corrosion cracking example AMP contained in Appendix B of NUREG-1927 Revision 1. The staff reviewed the scope of the AMP, areas of inspection coverage, inspection intervals, detection requirements, detection methods, and the monitoring and trending approach in the DSC external surfaces AMP, and determined that the AMP provides reasonable assurance that the DSC subcomponents will maintain their intended function through the renewal period. The applicant cited the general licensee's previously approved Quality Assurance Program for assuring that adequate corrective actions are taken if the acceptance criteria is not met. The applicant stated that the licensee's Corrective Action Program ensures that conditions adverse to quality are promptly identified and corrected, including root cause determinations and prevention of recurrence. The applicant noted that deficiencies are either corrected, or are evaluated as acceptable for continued service through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with current licensing basis conditions. The applicant stated that evaluations performed to assess conditions associated with aging need to follow the same methodology used in the licensing and design bases calculations. The staff finds that the existing approved Corrective Action Program for each general licensee is adequate for assuring these corrective actions are taken on a timely manner.

The applicant stated the detection of aging effects is performed using VT-3 and VT-1 visual inspection procedures consistent with the ASME Code Section XI, IWA-2200 (ASME, 2010) to monitor for material degradation of the DSC shell assembly. In addition, the applicant stated that the appearance and location of atmospheric deposits on the canister surfaces must be recoded. The applicant stated that the assessment of surface condition for the loss of material from pitting corrosion and crevice corrosion are to be assessed using visual inspection to the requirement of VT-1 or equivalent per ASME Code Section XI, IWA-2211 and the assessment of surface condition and cracks as a result of stress corrosion cracking are to be assessed using visual inspection to the requirement of VT-1 or equivalent per ASME Code Section XI, IWA-2211. The applicant stated that personnel performing visual examinations shall be qualified and certified in accordance with Code Section XI, IWA-2300, including the requirements of Section XI, Appendix VI.

The staff determined that VT-1 visual inspection identified in the parameters monitored or inspected is adequate for the detection of pitting corrosion. Crevice corrosion may also be detected provided the crevice area is accessible. Detection of cracks using visual methods is dependent on a number of factors (Cumblidge et al., 2004, 2007) including crack opening displacement, surface finish and lighting. However, the results of SCC testing (He et al., 2014)

have shown that pitting and the formation of corrosion products typically accompany atmospheric CISCC cracks. The staff note that operational experience has shown that corrosion products associated with iron contamination were readily identified during lead system inspection of a NUHOMS system (Waldrop et al., 2014). The staff finds the parameters monitored or inspected provides reasonable assurance for managing the aging mechanisms and effects identified in the AMR results for the DSC external surfaces.

The applicant stated that acceptance criteria are defined to ensure that the need for corrective actions will be identified before loss of intended functions. The applicant provided clear and specific acceptance criteria for VT-3 and VT-1 visual inspections. The applicant stated that acceptance of degraded condition for continued service is in accordance with facility procedural requirements, and includes an engineering evaluation performed using plant design procedures. The applicant also stated that acceptance of degraded condition for continued service is in accordance with industry codes and standards, and conforms to the CoC licensing basis. The applicant stated that evaluation for continued service of a DSC found to exhibit localized corrosion or cracking performed in accordance with ASME Code Section XI, IWB-3514.1 and IWB-3640 (ASME, 2010) or equivalent criteria is acceptable. The applicant identified that when visual examination detects evidence of localized corrosion, the affected areas will be further examined to determine the extent and the depth of penetration. The additional information that would be required is dependent on the nature of the defects but could include (1) surface-connected crack length, (2) surface-connected crack length and depth and (3) extent and depth of the corrosion.

The staff reviewed the applicant's acceptance criteria and determined that the criteria are acceptable because they require either the absence of degradation or an engineering evaluation to DSCs that are identified with aging effects. Further, the acceptance criteria are consistent with the guidance provided in NUREG-1927 Revision 1 Appendix B. With the inclusion of the specific acceptance criteria in the DSC external surfaces AMP, the staff concludes that signs of deterioration will be adequately detected and appropriately addressed before degradation reaches a level where the DSC subcomponent would be challenged in performing its intended function. The staff finds that the AMP Acceptance Criteria provides reasonable assurance for managing the aging mechanisms and effects identified in the AMR of the welded stainless steel DSCs.

The applicant stated that corrective actions are in accordance with site quality assurance (QA) procedures and review and approval processes, and that administrative controls are implemented according to the requirements of the general licensee's 10 CFR Part 50 Appendix B Program. The applicant further stated that deficiencies are either corrected, or are evaluated as acceptable for continued service through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with current licensing basis conditions. The applicant stated an extent of condition investigation, per the licensee's corrective action program, may trigger additional inspections via a different method, increased inspection frequency or expanded inspection sample size.

The staff finds that the corrective actions DSC External Surfaces AMP are acceptable because corrective actions are implemented in accordance with the licensee's Appendix B Program and require a root cause analysis, an evaluation to determine acceptability for continued service or correction of deficiencies to assure that intended functions are maintained.

The applicant stated that the DSC External Surfaces AMP is a "learning" AMP and the AMP will be updated, as necessary, to incorporate new information on degradation due to aging effects

identified from plant-specific inspection findings, related industry operating experience (OE), and related industry research. The applicant stated that future plant-specific and industry OE is captured through the licensee's OE review process following the regulatory framework for the consideration of OE concerning aging management and aging-related degradation in LR-ISG-2011-05 (NRC, 2011b). The applicant stated that ongoing review of both plant-specific and industry OE will continue through the period of extended operation to ensure that the DSC External Surfaces AMP continues to be effective in managing the identified aging effects. The applicant noted that reviews of OE by the licensee in the future may identify areas where this AMPs should be enhanced, updated, or if a new program should be developed.

The staff reviewed the applicant's description of the learning AMP including plans for periodic reviews of the AMP and updates as necessary. The staff have determined that periodic updates to incorporate new information on degradation due to aging effects identified from plant-specific inspection findings, related industry OE, and related industry research are appropriate and provides reasonable assurance that the AMP will continue to be effective throughout the period of certificate renewal. The staff finds that the DSC External Surfaces AMP provides reasonable assurance for managing the aging mechanisms and effects identified in the AMR of the welded stainless steel DSCs.

### 3.6.3 DSC Aging Management Program for the Effects of CISCC

The applicant has stated that the DSC AMP for the Effects of CISCC applies to CoC 1004 Amendments 0 through 11, 13, and 14 (when approved) and applies to ISFSI locations that may have sufficient atmospheric chlorides to initiate CISCC within the forty-year license extension period. The applicant noted that the DSC AMP for the Effects of CISCC is similar to the DSC External Surfaces AMP with the addition of surface deposit collection and analysis for chloride concentration, and the addition of NDE more sensitive than VT-1 to the recommended corrective actions.

The applicant identified the sequence of the DSC AMP for the Effects of CISCC as:

1. A plant-specific applicability evaluation is performed to determine if the CISCC aging effect is applicable to the licensee's ISFSI. The applicant cited NRC Information Notice IN 2012-20 (NRC, 2012), which identified environments near saltwater bodies or other sources of chlorides, such as salted roads (Allen and Stensland, 2006) or condensed cooling tower water as locations where stainless steel SSCs may have increased susceptibility to CISCC.

2. Baseline inspection comprises remote visual inspection of the DSC surface, and collection and analysis of wet and dry surface deposit samples.

3. When indications of aging degradation are detected by visual inspection, a more sensitive NDE method is recommended to quantify the degradation in order to evaluate for continued service and to ensure the DSC will continue to perform its intended function.

4. Inspection conditions deviating from the acceptance criteria are entered for resolution into the licensee's corrective action program. Such conditions are documented using approved processes and procedures, such that results can be trended and corrected.

The applicant stated that general licensees may use inspections results from other general licensee inspections if it can be demonstrated that the other general licensee inspections are

bounding. Parameters to be considered in making a bounding determination were identified by the applicant as: similar or more benign environmental conditions, similar storage system design components, similar stored fuel parameters, heat load, and operational history. The applicant stated that the criteria for DSC selection or the bounding determination shall be justified and documented in the general licensee's 10 CFR 72.212(b)(5) evaluation.

The staff reviewed the narrative included in the 10 elements of the DSC AMP for the Effects of CISCC and finds that the AMP follows the guidance provided in NUREG 1927 Revision 1 Section 3.6 and that the description of each AMP element is consistent with the Localized Corrosion and Stress corrosion cracking example AMP contained in Appendix B of NUREG 1927 Revision 1. The staff reviewed the scope of the AMP, areas of inspection coverage, inspection intervals, detection requirements, detection methods, and the monitoring and trending approach in the DSC AMP for the Effects of CISCC, and determined that the AMP provides reasonable assurance that the DSC subcomponents will maintain their intended function through the renewal period. The applicant cited the general licensee's previously approved Quality Assurance Program for assuring that adequate corrective actions are taken if the acceptance criteria are not met. The applicant stated that the licensee's Corrective Action Program ensures that conditions adverse to quality are promptly identified and corrected, including root cause determinations and prevention of recurrence. The applicant noted that stated that deficiencies are either corrected, or are evaluated as acceptable for continued service through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with current licensing basis conditions. The applicant stated that evaluations performed to assess conditions associated with aging need to follow the same methodology used in the licensing and design bases calculations. The staff finds that the existing approved Corrective Action Program for each general licensee is adequate for assuring these corrective actions are taken on a timely manner.

The applicant stated the detection of aging effects is performed using VT-3 and VT-1 visual inspection procedures consistent with the ASME Code Section XI, IWA-2200 (ASME, 2010) to monitor for material degradation of the DSC shell assembly. In addition, the applicant stated that the appearance and location of atmospheric deposits on the canister surfaces must be recoded. The applicant stated that the assessment of surface condition for the loss of material from pitting corrosion and crevice corrosion are to be assessed using visual inspection to the requirement of VT-1 or equivalent per ASME Code Section XI, IWA-2211 and the assessment of surface condition and cracks as a result of stress corrosion cracking are to be assessed using visual inspection to the requirement of VT-1 or equivalent per ASME Code Section XI, IWA-2211. The applicant stated that personnel performing visual examinations shall be qualified and certified in accordance with Code Section XI, IWA-2300, including the requirements of Section XI, Appendix VI.

The staff reviewed the applicant's detection of aging effects and determined that the visual inspection method and approach described in the DSC AMP for the Effects of CISCC is appropriate for detecting the effects of aging including localized corrosion such as pitting and crevice corrosion and stress corrosion cracking of the welded stainless steel DSCs. The staff reviewed the DSC AMP for the Effects of CISCC and determined that the methods used for the detection of aging effects are consistent with the guidance provided in NUREG-1927 Revision 1 Appendix B. The staff reviewed the DSC surface coverage proposed by the applicant in the DSC AMP for the Effects of CISCC and has determined that the areas of coverage are appropriate for detecting aging of the welded stainless steel DSCs including localized corrosion and CISCC. The staff reviewed inspection criteria provided by the applicant in the DSC AMP for the Effects of CISCC and determined that the graded approach indicated by the applicant, using

VT-3 and VT-1 inspection methods, is appropriate for detecting indications localized corrosion and evidence of the potential for CISCC on the welded stainless steel DSC surfaces. The staff also reviewed the personnel qualifications and has confirmed that the applicant confirmed that personnel performing visual examinations shall be qualified and certified in accordance with ASME Code Section XI, IWA-2300, including the requirements of Section XI, Appendix VI. The staff determined that these requirements are appropriate for personnel conducting visual examinations.

The applicant stated that acceptance criteria are defined to ensure that the need for corrective actions will be identified before loss of intended functions. The applicant provided clear and specific acceptance criteria for VT-3 and VT-1 visual inspections. The applicant stated that acceptance of degraded condition for continued service is in accordance with facility procedural requirements, and includes an engineering evaluation performed using plant design procedures, is in accordance with industry codes and standards, and conforms to the CoC licensing basis. The applicant stated that evaluation for continued service of a DSC found to exhibit localized corrosion or cracking performed in accordance with ASME Code Section XI, IWB-3514.1 and IWB-3640 (ASME, 2010) or equivalent criteria is acceptable. The applicant identified that when visual examination detects evidence of localized corrosion, the affected areas will be further examined to determine the extent and the depth of penetration. The additional information would be that required is dependent on the nature of the defects but could include (1) surface-connected crack length, (2) surface-connected crack length and depth and (3) extent and depth of the corrosion.

The staff reviewed the applicant's acceptance criteria and determined that the criteria are acceptable because they require either the absence to degradation or an engineering evaluation to DSCs that are identified with aging effects, and that the acceptance criteria are consistent with the guidance provided in NUREG-1927 Revision 1 Appendix B. With the inclusion of the specific acceptance criteria in the DSC external surfaces AMP, the staff concludes that signs of deterioration will be adequately detected and appropriately addressed before degradation reaches a level where the DSC subcomponent would be challenged in performing its intended function. The staff finds that the AMP Acceptance Criteria provides reasonable assurance for managing the aging mechanisms and effects identified in the AMR of the welded stainless steel DSCs.

The applicant stated that corrective actions are in accordance with site quality assurance (QA) procedures, review and approval processes, and that administrative controls are implemented according to the requirements of the general licensee's 10 CFR Part 50 Appendix B Program. The applicant further stated that deficiencies are either corrected or evaluated as acceptable for continued service through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with current licensing basis conditions. The applicant stated an extent of condition investigation, per the licensee's corrective action program, may trigger additional inspections via a different method, increased inspection frequency or expanded inspection sample size.

The staff finds that the corrective actions in the DSC AMP for the Effects of CISCC are acceptable because corrective actions are implemented in accordance with the licensee's Appendix B Program and require a root cause analysis, an evaluation to determine acceptability for continued service or correction of deficiencies to assure that intended functions are maintained.

The applicant stated that the DSC Aging Management Program for the Effects of CISCC is a "learning" AMP and the AMP will be updated, as necessary, to incorporate new information on degradation due to aging effects identified from plant-specific inspection findings, related industry operating experience (OE), and related industry research. The applicant stated that future plant-specific and industry OE is captured through the licensee's operational experience review process following the regulatory framework for the consideration of OE concerning aging management and aging-related degradation in LR-ISG-2011-05 (NRC, 2011b). The applicant stated that ongoing review of both plant-specific and industry OE will continue through the period of extended operation to ensure that the DSC External Surfaces AMP continues to be effective in managing the identified aging effects. The applicant noted that reviews of OE by the licensee in the future may identify areas where this AMPs should be enhanced, updated, or if a new program should be developed.

The applicant provided a description of the tollgates that will be used to evaluate and revise the DSC AMP for the Effects of CISCC. The initial tollgate is to perform initial inspection of selected DSCs as specified in in the DSC AMP for the Effects of CISCC and as updated at the time that planning for the inspection begins. The second tollgate occurs 5 years after the initial AMP is conducted. This tollgate is to evaluate information from information sources and perform a written assessment of the aggregate impact of the information, including but not limited to corrective actions required and the effectiveness of the DSC AMP for the Effects of CISCC. Tollgate 3 requires the GL to evaluate additional information gained from the sources listed in tollgate 2 along with any new relevant sources and perform a written assessment of the aggregate impact of the information, including results of tollgate 2. The age related degradation mechanisms evaluated at this tollgate and the time at which it is conducted may be adjusted based on the results of the tollgate 2. Subsequent tollgates are informed by the information gained in the preceding tollgate.

The staff reviewed the applicant's description of the tollgates for the DSC AMP for the Effects of CISCC. The staff find that the applicant has identified relevant information sources that should be reviewed as part of evaluating the effectiveness and, if necessary, revising the DSC AMP for the Effects of CISCC. Relevant research is currently being conducted by both DOE and EPRI including an assessment of the susceptibility factors for CISCC. EPRI has also recently evaluated the conditions for CISCC (Gorman et al., 2014) and evaluated failure modes and their effects (Fuhr et al., 2013) for welded stainless steel storage canisters. The Central Research Institute of Electric Power Industry (CRIEPI) in Japan has been conducting testing and evaluating atmospheric factors that influence CISCC since the mid 1980's. As CISCC is not unique aging mechanism for spent fuel storage systems, a review of relevant foreign and domestic nuclear and non-nuclear operating experience is appropriate. Information gained from the inspection of DSC located in a range of environments will provide valuable information that can be used to inform and update the DSC AMP for the Effects of CISCC. The staff find that the information sources referenced in the renewal application and the actions identified at each tollgate provides reasonable assurance that the DSC AMP for the Effects of CISCC will be continue to be effective for managing CISCC as a potential aging mechanism for the DSC.

### 3.6.4 Horizontal Storage Module Aging Management Program for External and Internal Surfaces

The applicant credited the "HSM Aging Management Program for External and Internal Surfaces" described in Appendix 6A of the renewal application for managing the aging mechanisms and effects of the horizontal storage modules (HSMs) and the concrete pad

(basemat). The applicant stated that this AMP applies to amendments 0 through 11, 13, and 14.

The staff reviewed the description in the 10 elements of the High Burnup Fuel AMP, and finds that the applicant's AMP provided the information expected per the example AMP in Table B-2 of NUREG-1927, Revision 1. More specifically, the applicant adequately described the activities to be performed as part of the program, the aging effects and mechanisms being managed, parameters to be monitored and inspected, methods and scope of areas to be inspected or monitored, and monitoring and trending approach. The applicant also defined an acceptance criteria consistent with that in Table B-2 of NUREG-1927, Revision 1 (NRC, 2016) and ACI 349.3R-02 (ACI, 2002). The applicant cited the general licensee's previously approved Quality Assurance Program for assuring that adequate corrective actions are taken if the acceptance criteria are not met. The staff finds that the existing approved Corrective Action Program for each general licensee is adequate for assuring these corrective actions are taken on a timely manner.

The applicant stated that the first examination will be generally performed on the selected HSM(s) at each site prior to entering the period of extended storage. However, the applicant requested a graduated grace period for general licensees who loaded early in the initial CoC licensed period. The applicant justified the grace period by clarifying that adequate time will be required for licensees to develop detailed inspection procedures, acquire tooling, and complete personnel training to perform remote non-destructive evaluation of the HSMs. The applicant identified the timing of the first (baseline) inspections in Table 6A-3 of the renewal application. The staff reviewed the cited operating experience and finds that the graduated schedule is justified, because it is commensurate with the timing of the inspections, as described standard guide ACI 349.3R-02 (ACI, 2002), which provides the consensus expert criteria used in the development of the example AMP in Table B-2 of NUREG-1927, Revision 1 (NRC, 2016). The staff, therefore, finds that the applicant provided an adequate AMP for managing the aging effects in approved Horizontal Storage Modules and general-licensee constructed storage pads (basemats), which the staff evaluated to be consistent with the guidance in Appendix B of NUREG-1927, Revision 1 (NRC, 2016).

### 3.6.5 HSM Inlets and Outlets Ventilation Aging Management Program

The applicant identified the "HSM Inlets and Outlets Ventilation Aging Management Program" described in Appendix 6A of the renewal application for managing the effects from reduced heat convection capacity due to HSM inlet and outlet vents blockage. The applicant stated that the purpose of the AMP is to ensure that the air ventilation inlets and outlets components (bird screens) are maintained free from blockage.

The staff reviewed the AMP per the guidance in NUREG-1927, Revision 1 (NRC, 2016). The staff's evaluation of each of the program elements is as follows:

1. Scope of Program

The applicant stated that the scope of the program consists of daily visual walkdown inspections or daily HSM temperature measurements to ensure that the air inlets and outlets are maintained free from blockage. Maintaining the inlets and outlets free from obstruction ensures temperatures are not elevated for prolonged periods, the concrete is not subject to high temperature damage, and fuel cladding temperatures are maintained within design bases limits. The applicant clarified that the activities under this AMP apply to all the HSMs installed at a site.

The applicant identified the following aging effects and mechanisms to be managed by this AMP:

- Reduced heat convection capacity due to HSM inlet and outlet vents blockage

The staff reviewed the Scope of Program and determined that the applicant adequately defined the activities to performed, the subcomponents within the scope of the AMP, and the aging mechanisms and effects to be managed by the program. The staff notes that the identified effects are not age-related, but event-driven. The activities in the program ensure that off-normal conditions are prevented, which could result in thermally-induced degradation of the HSM concrete components and/or the fuel cladding. Therefore, this program assures that the design-bases temperature limits are maintained, which provide assurance that the spent fuel configuration will be maintained during the period of extended operation, as well as preventing adverse degradation of the DSS components. The staff finds the Scope of Program provides reasonable assurance that the effects identified will be adequately managed by the activities within the program.

2. Preventive Actions

The applicant stated that the program is a condition monitoring program. No preventive actions beyond those defined by the activities in this program need be performed.

The staff reviewed the Preventive Actions and finds that the AMP does not require preventive actions to provide effective aging management. The staff finds that the applicant adequately concluded that preventive actions are not required for this program.

3. Parameters Monitored or Inspected

The applicant stated that the HSM inlet and outlet vents are monitored by visual inspection or daily HSM temperature measurements to ensure that temperatures are not elevated for prolonged periods due to blockage of the inlet and outlet vents of the HSM. The applicant stated that this measure prevents thermally induced damage to concrete components and the fuel cladding.

The staff reviewed the applicant's Parameters Monitored or Inspected for the visual inspections of the HSM inlet and outlet vents. The staff finds that the Parameters Monitored or Inspected provide reasonable assurance for managing the identified effects, and ensuring the intended functions of the HSM will be maintained during the period of extended operation, because the Parameters Monitored or Inspected will prevent temperatures exceeding design-basis limits.

4. Detection of Aging Effects

The applicant stated that the program manages reduction of heat transfer capability due to blockage of air inlet and outlet openings using visual inspection or daily HSM temperature measurements. The applicant further stated that visual inspections should be performed daily and should cover 100% of the accessible components to ensure that the components' intended function is maintained. This activity has been required since the initial period of operation per a CoC technical specification.

The applicant stated that, if the visual inspection method is used, a daily visual inspection of the HSM air inlets and outlets is performed to ensure that no material accumulation blocks the airflow. Any unusual conditions or blockages are recorded.

The daily HSM temperature measurements could be any parameter such as (1) a direct measurement of the HSM temperatures, (2) a direct measurement of the DSC temperatures, (3) a comparison of the inlet and outlet temperature difference to predicted temperature differences for each individual HSM, or (4) other means that would identify and allow for the correction of off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria.

The staff reviewed the inspection methods and sample sizes for the HSM inlets and outlets, and finds these provide acceptable means to effectively detect the effects identified in the scope of the program. The staff finds that the Detection of Aging Effects provides reasonable assurance that the HSM subcomponents will maintain their intended functions for the period of extended operation.

5. Monitoring and Trending

The applicant stated that visual inspections are performed daily and associated personnel are qualified in accordance with site-controlled procedures and processes as prescribed in technical specifications. The applicant further stated that monitoring and trending activities are used to track degradation. The applicant clarified that deficiencies are documented using approved processes and procedures, such that results can be trended and corrected. This monitoring is conducted in accordance with the provisions of 10 CFR Part 50, Appendix B.

The staff reviewed the applicant's Monitoring and Trending for the visual inspections of the HSM inlets and outlets. The staff expects all monitoring and trending activities under this AMP will be commensurate with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" or 10 CFR 72, Subpart G, "Quality Assurance," as applicable. The staff determines that the applicant's description of monitoring and trending methods provides acceptable means to effectively predict the extent of the degradation and timely corrective actions. The staff determined that the monitoring and trending methods for the HSM subcomponents provide acceptable means to effectively detect the aging mechanisms and effects in the AMR and that, therefore, the HSM subcomponents will maintain their intended functions for the period of extended operation.

6. Acceptance Criteria

The applicant stated that the acceptance criteria is the condition where inlet and outlet vents are free from blockage. The acceptance criteria for temperature measurements is that measured concrete temperatures are below the short-term concrete temperature limits for blocked vent condition in the UFSAR.

The staff reviewed the applicant's Acceptance Criteria for the visual inspections of the HSM inlet and outlet vents. The staff finds that the Acceptance Criteria provides reasonable assurance that the HSM subcomponents will maintain their intended functions for the period of extended operation because the activities will prevent design-bases temperature limits from being exceeded.

7. Corrective Actions

The applicant stated that site-specific quality assurance procedures, review and approval processes, and administrative controls are implemented according to the requirements of 10 CFR Part 50 Appendix B. The applicant further stated that, as part of their quality assurance procedures, air vents (inlets or outlets) shall be cleared in the event that the surveillance shows

blockage. If the bird screen is damaged, it shall be replaced. The applicant stated that the licensee's corrective action program ensures that conditions adverse to quality are promptly identified and corrected, including root cause determinations and prevention of recurrence. The applicant clarified that deficiencies are either corrected or are evaluated to be acceptable for continued service through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with the design bases. The applicant further clarified that an extent of condition investigation per the licensee's corrective action program may trigger additional inspections via a different method, increased inspection frequency and/or expanded inspection sample size.

The staff expects that all corrective actions under AMP will be commensurate with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," because the applicant referenced the licensee's correction action program, which meets the quality assurance requirements in 10 CFR 50 Appendix B. Therefore, the applicant's corrective action procedures provide reasonable assurance that corrective actions will be adequate for managing the aging mechanisms and effects identified in the AMR of the HSM subcomponents.

8. Confirmation Process

The applicant stated that confirmatory actions, as needed, are implemented as part of each general licensee's approved Corrective Action Program (CAP).

The staff reviewed the details provided for the Confirmation Process, in the context of the existing Quality Assurance (QA) Program, to ensure that appropriate corrective actions are completed and are effective. The existing approved QA Program has proven adequate for the same activities under this AMP, which have been conducted to date under the initial storage period. Therefore, the staff expects that the QA Program will continue to be effective during the period of extended operation. The staff concludes that each general licensee's CAP provides reasonable assurance that the Confirmation Process is adequate for managing the aging mechanisms and effects identified in the AMR of the HSM subcomponents.

9. Administrative Controls

The applicant stated that administrative controls under the general 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 period of extended operation. General licensees use the regulatory requirements in 10 CFR 72.74, 72.75, and 72.242 to determine if a particular aging-related degradation condition or event identified via operating experience, research, monitoring, or inspection is reportable to the NRC. The applicant clarified that 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 applicant as outlined in Section 2.6.6 of NEI 14-03, Rev. 0 (2014).

The staff reviewed the details provided for administrative controls, in the context of the existing Quality Assurance Program, to ensure that the administrative controls will be adequate to provide a formal review and approval process. The staff concludes that the general licensee's Quality Assurance Program, per the requirements in 10 CFR 50 Appendix B, provides reasonable assurance that the Administrative Controls are adequate for managing the aging mechanisms and effects identified in the AMR of the HSM subcomponents.

10. Operating Experience

The applicant stated that visual surveillance of the exterior of the air inlets and outlets and inspection ventilation blockage is a required technical specification requirement for the NUHOMS® HSM System for both the initial storage period and the period of extended operation. Consequently, substantial operating experience exists for these activities. As an alternative to the daily surveillance of the HSM air ventilation system, a daily temperature measurement of each HSM may be conducted per the technical specification requirement.

The applicant further stated that the "HSM Inlets and Outlets Ventilation Aging Management Program" ensures that HSM air inlets and outlets are not blocked for more than the analyzed time period to prevent exceeding the allowable HSM concrete temperatures or the fuel cladding temperatures.

The applicant identified experience of partial blockage of HSM air inlet duct screens from snowfall. The applicant further stated that heat stored in and by the HSM thermal mass quickly melts any partial "snow build-ups" after the snowfall ceases. The staff concludes that the applicant's operating experience demonstrates that the existing surveillance/maintenance activity has functioned effectively during the initial storage period. Therefore, this operating experience supports the staff's conclusion that the AMP will continue to function effectively in the period of extended operation.

Learning AMP

The applicant stated that the "HSM Inlet and Outlets Ventilation Aging Management Program" is a "learning" AMP. The applicant clarified that this means that the AMP will be updated, as necessary, to incorporate new information on degradation due to aging effects identified from plant-specific inspection findings, related industry operating experience (OE), and related industry research.

The applicant further stated that future plant-specific and industry OE is captured through the licensee's review process of OE, following the regulatory framework for the consideration of OE concerning aging management and aging-related degradation in LR-ISG-2011-05 (NRC,2011b). The ongoing review of both plant-specific and industry OE will continue through the period of extended operation to ensure that the program continues to be effective in managing the identified aging effects. The reviews of OE by the licensee in the future may identify areas where AMPs should be enhanced or new programs developed.

The applicant further stated that the CoC holder and the general licensees are to maintain the effectiveness of this AMP under their respective QA programs used to meet the criteria of 10 CFR Part 72, Subpart G, and 10 CFR Part 50 Appendix B, respectively.

The staff reviewed OE cited by the applicant and did not find events that would indicate that the "HSM Inlet and Outlets Ventilation Aging Management Program" would not be effective in managing the effects identified in the scope of the program. Therefore, the staff finds that the discussion on OE in the renewal application provides reasonable assurance that the AMP will be adequate for managing the aging mechanisms and effects identified in the AMR of the HSM.

### 3.6.6 Transfer Cask Aging Management Program

The applicant described the Transfer Cask Aging Management Program in Section 6A.7 of the renewal application and stated that the AMP applies to CoC 1004 Amendments 0 through 11, 13 and 14. The applicant stated that the program manages loss of material due to general corrosion, crevice corrosion, pitting corrosion, and wear of SSCs through visual inspections and liquid penetrant examinations (trunnion bearing surfaces and welds only).

The staff reviewed each of the elements of the AMP with respect to the guidance in NUREG-1927 to ensure that the program is capable of adequately managing the effects of aging.

1. Scope of Program

The applicant stated that the scope of the program includes visual inspection and monitoring of all accessible TC subcomponents for loss of material due to general, pitting, and crevice corrosion, and loss of material due to wear. In addition, the program includes dye penetrant testing on trunnion surfaces to identify wear. The applicant also stated that the AMP focuses on

- stainless steel subcomponents that are intermittently exposed internally to the liquid (water) neutron shield and external surfaces exposed to outdoor and indoor conditions

- carbon steel subcomponents that are exposed to weather or other forms of moisture

Specifically, the program includes ASME Code Section XI, VT-3 visual inspections of all accessible surfaces, including those in the cask cavity and on OS197L supplemental shielding. If any area of degradation is found during these inspections, they will be further examined with VT-1 visual inspections. Existing routine maintenance activities may be credited towards this AMP, provided that the activities meet the AMP requirements and acceptance criteria.

The staff reviewed the scope of program with respect to the results of the applicant's AMR and the staff's evaluation of the identified aging mechanisms and effects documented above in Section 3.3.5.2 of this SER. The staff notes that the applicant identified the specific transfer cask materials and subcomponents covered by the AMP and the aging mechanisms and effects to be managed, consistent with the guidance in NUREG-1927. Therefore, the staff finds that the Scope of Program provides reasonable assurance that the effects of aging will be adequately managed.

2. Preventive Actions

The applicant stated that the Transfer Cask Aging Management Program is a condition-monitoring program. However, the applicant also stated that preventive actions also shall include:

- The use of demineralized water in the liquid neutron shield to prevent deterioration of the shield jacket
- Draining of the liquid neutron shield before storage to prevent damage due to freezing and to mitigate corrosion,

and should include:

- Storage of the TC in a manner to prevent exposure to precipitation

- Prevention of contact between the surface of the TC and tarpaulins to prevent accumulation of condensation
- Rinsing of the TC with demineralized water as it is raised out of the spent fuel pool

The staff reviewed the preventive actions and determined that the guidance for minimizing the transfer casks' exposure to environmental moisture and contaminants are effective mitigation activities to prevent loss of material. The staff notes that damage due to freezing of the liquid neutron shield was not an aging effect identified by the applicant in the AMR. However, the staff notes that draining removes the possibility for damage due to expansion of the liquid neutron shield.

The staff finds that the Preventive Actions program element provides reasonable assurance that the effects of aging will be adequately managed because the applicant described the activities that will prevent or mitigate the effects of aging, consistent with the guidance in NUREG-1927 .

3. Parameters Monitored or Inspected

The applicant stated that the parameters inspected by the program include visual evidence of degradation of external surfaces of the transfer cask, trunnions, fasteners, and supplemental shielding. Specifically, all cask surfaces are inspected for signs of deterioration (e.g., corrosion, signs of liquid neutron shield leakage). The trunnions, inner liner, TC rails, and fasteners are also inspected for indications of damage and wear.

The staff reviewed the parameters monitored or inspected and determined that the specific parameters that will be monitored and inspected are identified and are capable of detecting loss of material of the transfer cask subcomponents. The staff also notes that the parameters provide a clear link to the aging effects identified in the AMR and managed by the program. Therefore, the staff finds that the Parameters Monitored or Inspected program element provides reasonable assurance that the effects of aging will be adequately managed.

4. Detection of Aging Effects

The applicant stated that the program visually inspects the accessible surfaces of the transfer cask, including the trunnions, the cask lid, and the OS197 supplemental shielding for loss of material every five years or prior to a fuel loading campaign at a given site. These visual inspections for corrosion and wear are performed in accordance with VT-3 examinations set forth in ASME Code Section XI, IWA-2213. The staff notes that the ASME Code states that VT-3 examinations are conducted to determine the general mechanical and structural condition of components. When the external surfaces of the TC neutron shield exhibit evidence of degradation, ASME Code VT-1 examinations per IWA-2211 are performed. Fasteners are also inspected for thread condition, corrosion, wear, or other degradation. Visual inspection personnel are to be qualified in accordance with the ASME Code Section XI, IWA-2300.

The applicant also stated that the program performs visual and liquid penetrant examinations on the upper and lower trunnion bearing surfaces and the accessible trunnion structural welds once every five years. The penetrant examinations are performed in accordance with ASME Code Section III, NC-5350.

The applicant further stated that existing routine maintenance requirements described in the UFSAR can be credited towards the AMP activities, provided that the maintenance activities are performed to the same requirements and acceptance criteria as set forth in this program.

Finally, the applicant stated that, although coatings are not credited for the prevention of corrosion of carbon steel components, the coatings on the OS1927L TC supplemental shielding will be managed to ensure the thermal performance of the shielding. As the coating is an integral part of the supplemental shielding, they will be evaluated as part of the VT-1 and VT-3 inspections described above, but with specific coating acceptance criteria defined in the Acceptance Criteria program element.

The staff reviewed the activities associated with the detection of aging effects to ensure that they are sufficient to identify degradation prior to a loss of intended function. The staff notes that the use of ASME Code Section XI visual inspection criteria is consistent with the recommendations in NUREG-1927 for the inspection of metallic canister components for corrosion. Similarly, the performance of dye penetrant examinations in accordance to the ASME Code is considered to provide sufficient controls to ensure that degradation of trunnions surfaces will be identified. Also, given that transfer casks are typically housed indoors, and the fact that the casks are subject to routine maintenance examinations prior to use, the minimum 5-year inspection interval is considered capable of identifying degradation prior to a loss of function.

The staff notes that, as detailed above, the performance of consensus ASME Code examinations on a minimum 5-year frequency is capable of identifying degradation prior to a loss of function. Also, the applicant provided sufficient detail to describe these examinations such that general licensees can appropriately implement the program. Therefore, the staff finds that the Detection of Aging Effects program element provides reasonable assurance that the effects of aging will be adequately managed.

5. Monitoring and Trending

The applicant stated that a baseline inspection is performed as part of the monitoring and trending activities, and the results of the baseline inspection are used for subsequent trending. The applicant also stated that deficiencies are documented using approved processes and procedures to allow for trending and corrective actions. This activity is performed in accordance with the CoC holder's 10 CFR Subpart G Program or the general licensee's 10 CFR Part 50 Appendix B Program.

The staff reviewed the applicant's monitoring and trending activities to ensure that they provide for an evaluation of the extent of aging and the need for timely corrective or mitigative actions. The staff notes that the performance of a baseline inspection and trending future inspection results against that baseline are consistent with the guidance in NUREG-1927 for effectively evaluating and responding to identified effects of aging. Thus, activities will be in place to ensure that the rate of degradation will be adequately evaluated such that future inspections will be performed, or components will be repaired, prior to a loss of functions. Therefore, the staff finds that the Monitoring and Trending program element provides reasonable assurance that the effects of aging will be adequately managed.

6. Acceptance Criteria

The applicant stated that the acceptance criteria for the Transfer Cask AMP includes the following:

- For all transfer cask subcomponents:
    - No corrosion of any subcomponent or wear of the inner liner

- For trunnion bearing surfaces and accessible welds:
    - Liquid penetrant examinations are evaluated against ASME Code Section III, NC-5350 acceptance criteria. The staff notes that the ASME Code criteria states that any linear indications greater than 1.5mm in length are unacceptable.
- For coatings of the OS197L supplemental shielding:
    - Coating condition is evaluated using ASTM D7167-12, Section 10.2 as a guide. Peeling, delamination, blisters, cracking, flaking, rusting, and physical damage will be evaluated.

Any indications of corrosion or wear, as well as conditions not meeting the above liquid penetrant or coating criteria, are entered into the CoC holder's or licensee's corrective actions program.

The staff reviewed the applicant's acceptance criteria to verify that they provide specific benchmarks to prompt corrective actions prior to a loss of intended functions. The staff notes that the criteria are consistent with the parameters monitored, and they are capable of being detected using the inspection methods detailed in the Detection of Aging Effects program element. Also, establishing a threshold as the absence of corrosion or wear, or specific thresholds defined by consensus codes and standards is capable of ensuring that potential signs of aging will be appropriately evaluated and addressed prior to a loss of function. Therefore, the staff finds that the Acceptance Criteria program element provides reasonable assurance that the effects of aging will be adequately managed.

7. Corrective Actions

The applicant stated that corrective actions are performed in accordance with the CoC holder's or licensee's corrective action program, as applicable. The applicant also stated that Quality Assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the CoC holder's 10 CFR Part 72 Subpart G Program or the general licensee's 10 CFR Part 50 Appendix B Program. The applicant further stated that the corrective action program ensures that conditions adverse to quality are promptly identified and corrected, including root cause determination and prevention of recurrence.

NUREG-1927 states that an applicant may reference the use of a Corrective Actions Program approved under 10 CFR Part 72 Subpart G or 10 CFR Part 50 Appendix B. The staff finds that the use of these NRC-approved programs provides reasonable assurance that appropriate measures will be taken when aging issues arise and when the AMP acceptance criteria are not met. The staff also notes that the aging issues associated with the transfer cask (e.g., corrosion and wear of metallic components) are well-known degradation mechanisms in nuclear industry and licensee's existing programs to address these mechanisms are considered capable to addressing them for the transfer cask as well. Therefore, the staff finds that the Corrective Actions program element provides reasonable assurance that the effects of aging will be adequately managed.

8. Confirmation Process

The applicant stated that the confirmatory actions are performed in accordance with the CoC holder's or licensee's corrective action program (CAP), as applicable. The applicant stated that the CAP implements the QA program procedures, review and approval processes, and

administrative controls per the requirements of 10 CFR Part 72 Subpart G or 10 CFR Part 50 Appendix B.

The staff notes that NRC-approved QA programs are an accepted approach to ensure that the effectiveness of corrective actions are verified and that adverse trends will be monitored to address potential degradation prior to a loss of function. Therefore, the staff finds that the Confirmation Process program element provides reasonable assurance that the effects of aging will be adequately managed.

9. Administrative Controls

The applicant stated that administrative controls under the CoC holder's or licensee's QA procedures and corrective action program are implemented in accordance with the requirements of 10 CFR Part 50 Appendix B. The staff notes that quality assurance program required by 10 CFR Part 50 Appendix B includes controls for document issuance (e.g. procedures, drawings), records retention, inspection, and maintenance of measuring and testing equipment.

The applicant also stated that the 10 CFR Part 72 regulatory requirements will be used determine if particular age-related degradation conditions are reportable to the NRC. Events and conditions not reportable to the NRC are communicated to the CoC holder as outlined in NEI 14-03 (NEI, 2015). The staff notes that NEI 14-03 describes a general framework by which operating experience at a particular site is evaluated to determine whether it has broader applicability and thus should be shared with other sites. The staff also noted that NRC-approved 10 CFR Part 50 Appendix B programs are an accepted approach to provide for adequate review, approval, and fulfillment of activities that ensure SSCs will continue to perform satisfactorily in service. Therefore, the staff finds that the Administrative Controls program element provides reasonable assurance that the effects of aging will be adequately managed.

10. Operating Experience

The applicant provided a summary of transfer cask operating experience in a proprietary portion of the renewal application. These observations were gathered during the existing periodic maintenance inspections. The applicant stated that the continued use of the transfer casks is evidence of the effectiveness of existing inspections and activities in maintaining their condition and functionality.

<u>Learning AMP</u>

The applicant also stated that the AMP will be updated to incorporate new information on degradation due to aging effects identified from plant-specific and industry operating experience as well as industry research. The applicant further stated that the operating experience reviews will follow the NRC guidance for power reactor license renewal in LR-ISG-2011-05, "Ongoing Review of Operation Experience" (NRC, 2011b).

The staff reviewed the Operating Experience program element to ensure that past operating experience was appropriately considered and that the program includes provisions to conduct future reviews of operating experience to confirm the program's continued effectiveness. The staff notes that the degradation described in the applicant's proprietary operating experience review is effectively addressed by the proposed AMP activities. Specifically, the staff verified that the proposed visual inspection methods and frequency provide for timely identification of the observed effects of aging. Also, the staff notes that the NRC guidance in LR-ISG-2011-05

states that licensees should commit to future reviews of plant-specific and industry operating experience to determine whether AMPs need to be enhanced. In addition, the guidance states that licensees should evaluate plant-specific operating experience to determine whether it should be reported to the wider industry. As a result, the staff determined that the applicant provided sufficient prior operating experience to support effectiveness of the Transfer Cask AMP activities while also providing a framework for future operating experience reviews to ensure that AMPs will be adjusted as knowledge becomes available from new analyses, experiments, and inspection activities. Therefore, the staff finds that the Operating Experience program element provides reasonable assurance that the effects of aging will be adequately managed.

### 3.6.7 Evaluation Findings

The staff reviewed the aging management programs provided in the renewal application. The staff performed its review following the guidance provided in NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel" and relevant ISGs, as identified in Table 1.6-1. Based on its review, the staff finds:

F3.4 The applicant has identified programs that provide reasonable assurance that aging effects will be managed effectively during the period of extended operation, in accordance with 10 CFR 72.240(c)(3).

# 4 CHANGES TO CERTIFICATE OF COMPLIANCE AND TECHNICAL SPECIFICATIONS

This section provides a consolidated list of the changes to the CoC conditions and technical specifications resulting from the review of the renewal application, some which have been described throughout the previous sections of this SER. This section also describes the different amendments to which the changes apply since some changes do not apply to all of the CoC amendments. The basis of the changes is provided here for those changes that are not described elsewhere in this SER.

## CHANGES TO CERTIFICATE OF COMPLIANCE

1. **Added the following condition to the Initial CoC (Amendment 0) and Amendments 1 through 11, 13 and 14:**

   *"FSAR UPDATE FOR RENEWED COC*

   *The CoC holder shall submit an updated FSAR to the Commission, in accordance with 10 CFR 72.4, within 90 days of the effective date of the renewal. The updated FSAR 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 FSAR pursuant to the requirements of 10 CFR 72.248."*

The CoC holder has indicated that changes will be made to the UFSAR to address aging management activities resulting from the renewal of the CoC. This condition ensures that the UFSAR changes are made in a timely fashion to enable general licensees using the storage system during the period of extended operation to develop and implement necessary procedures.

2. **Added the following condition to the Initial CoC (Amendment 0) and Amendments 1 through 11, 13 and 14:**

   *"72.212 EVALUATIONS FOR RENEWED COC USE*

   *Any general licensee that initiates spent fuel dry storage operations with the Standardized NUHOMS® Horizontal Modular Storage System after the effective date of the renewal of the CoC and any general licensee operating a Standardized NUHOMS® Horizontal Modular Storage System as of the effective date of the renewal of the CoC, 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 FSAR 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 staff considers it important to ensure that appropriate considerations for the period of extended operation are evaluated in the general licensee's 72.212 report. These considerations arise from the analyses and assumptions in the renewal application regarding operations during the period of extended operation. This includes potential use by general licensees that may use new Standardized NUHOMS® Horizontal Modular Storage System after the CoC has been renewed whether at a new or at an existing general license ISFSI. The renewal of the CoC is based on assumptions and analyses regarding the dry storage system and the sites where it is used that licensees considering the use of the Standardized NUHOMS® Horizontal Modular Storage System should evaluate for use at their respective sites. This condition also makes it clear that general licensees that currently use a Standardized NUHOMS® Horizontal Modular Storage System will need to update their 10 CFR 72.212 reports even if they do not put additional dry storage systems into service after the renewal's effective date in order to meet the requirements in 10 CFR 72.212(b)(11).

**3. Added the following condition to the Initial CoC (Amendment 0) and Amendments 1 through 11, 13 and 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 assure they remain adequate for any changes to SSCs within the scope of renewal."*

The CoC may continue to be amended after it has been renewed. This condition ensures that amendments to the CoC also address aging management impacts that may arise from the changes to the system in proposed amendments.

**4. Revised Initial CoC (Amendment 0) and Amendments 1 through 11, 13 and 14 to address change to language in 10 CFR 72.210 and other updates to the regulations:**

This change is made for consistency with the language currently in 10 CFR 72.210 and other cited regulations, and is not pertinent to the safety review conducted for the renewal application. For expediency, the pertinent CoC revisions and associated Technical Specifications have been addressed as part of the CoC renewal. The following is an example of the change from CoC Amendment 1, Revision 1. Changes to the text are in bold, which only involves adding new text.

"Casks authorized by this certificate are hereby approved for use by holders of 10 CFR Part 50 **and 10 CFR Part 52** licenses for nuclear power 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 technical specifications."

Any CoC and technical specification language that discusses 10 CFR Part 50 licensees was modified to also include 10 CFR Part 52 licensees. In addition, updates to the regulation citations referenced in the applicable CoC and technical specifications have been changed to reflect citations currently in the regulations.

## CHANGES TO TECHNICAL SPECIFICATIONS

1. **Revised Technical Specification (TS) 1.1.2, "Operating Procedures," associated with the Initial CoC (Amendment 0) and Amendments 1 through 10, as follows:**

   Added the following new header and paragraphs to the end of TS 1.1.2:

   *Operating and Aging Management Program Procedures and Reporting*

   *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. Each procedure shall contain a reference to the specific aspect of the AMP element implemented by that procedure, and that reference shall be maintained even if the procedure is modified.*

   *The general licensee shall establish and implement these written procedures within 180 days of the effective date of the renewal of the CoC or 180 days of the 20th anniversary of the loading of the first dry storage system at its site, whichever is later. The general licensee shall maintain these written procedures for as long as the general licensee continues to operate Standardized NUHOMS® Horizontal Modular Storage Systems in service for longer than 20 years.*

   The above TS revision is similar in nature to the currently existing CoC conditions and TS regarding operating procedures for loading and operating dry storage systems under this CoC and extends the requirement for operating procedures to address AMP activities. The timeframe in the condition is to ensure operating procedures are developed in a timely manner and is consistent with conditions placed in specific licenses that have been renewed. The tollgate assessments in the AMPs assure that AMP procedures will be informed by the results of those assessments.

   The staff notes that the applicant proposed the above technical specification, which the staff modified to include an AMP implementation timeframe and require AMP referencing in the implementation procedures.

2. **Revised TS 1.2.4 (Bases paragraph, first two sentences) associated with the Initial CoC (Amendment 0) and Amendments 1 through 10, as follows:**

   *If the DSC leaked at the maximum acceptable rate of $1.0 \times 10^{-4}$ atm $cm^3/s$ for a period of 60 years, about 189,600 cc of helium would escape from the DSC. This is about 3.25% of the $5.83 \times 10^6$ $cm^3$ of helium initially introduced in the DSC.*

   The above TS revision was proposed by the applicant, which revises the analyzed period to 60 years, consistent with the initial approved period (20 years) added to the renewed period (40 years).

3. **Revised TS 5.1, "Procedures," associated with the Amendments 11, 13 and 14:**

   Added the following new subsection:

   *5.1.3 Aging Management Program Procedures and Reporting*

   *Each general licensee shall have a program to establish, implement, and maintain written procedures for each AMP described in the 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. Each procedure shall contain a reference to the specific aspect of the AMP element implemented by that procedure, and that reference shall be maintained even if the procedure is modified.*

   *The general licensee shall establish and implement these written procedures within 180 days of the effective date of the renewal of the CoC or 180 days of the 20th anniversary of the loading of the first dry storage system at its site, whichever is later. The general licensee shall maintain these written procedures for as long as the general licensee continues to operate Standardized NUHOMS® Horizontal Modular Storage Systems in service for longer than 20 years.*

   The above TS revision is similar in nature to the currently existing CoC conditions and TS regarding operating procedures for loading and operating dry storage systems under this CoC and extends the requirement for operating procedures to address AMP activities. The timeframe in the condition is to ensure operating procedures are developed in a timely manner and is consistent with conditions placed in specific licenses that have been renewed. The tollgate assessments in the AMPs assure that AMP procedures will be informed by the results of those assessments.

   The staff notes that the applicant proposed the above technical specification, which the staff modified to include an AMP implementation timeframe and require AMP referencing in the implementation procedures.

# 5 CONCLUSION

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 72.240(d), the design of a spent fuel storage cask will be renewed if (1) the quality assurance requirements in 10 CFR subpart G are met, (2) the requirements of 10 CFR 72.236(a) through (i) are met, and (3) the application includes a demonstration that the storage of spent fuel has not, in a significant manner, adversely affected structures, systems, and components important to safety. Additionally, 10 CFR 72.240(c) requires that the safety analysis report accompanying the application contain time-limited aging analyses and aging management programs that demonstrate that the dry storage system SSCs will continue to perform their intended functions for the requested period of extended operation.

The staff of the U.S. Nuclear Regulatory Commission (NRC) (the staff) reviewed the renewal application for the Standardized NUHOMS Storage System, in accordance with NRC regulations 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste." The staff followed the guidance provided in NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," and interim staff guidance (ISG) as identified in Table 1.6-1. Based on its review of the renewal application and the CoC conditions, the staff determines that the dry storage system has met the requirements of 10 CFR 72.240.

# 6 REFERENCES


ACI 215R-74, “Considerations for Design of Concrete Structures Subjected to Fatigue Loading.” Farmington Hills, Michigan: American Concrete Institute. 1997.

ACI 201.1R-08, “Guide for Making a Condition Survey for Concrete in Service.” Farmington Hills, Michigan: American Concrete Institute. 2008a.

ACI 209R-92, “Prediction of Creep, Shrinkage, and Temperature Effects in Concrete Structures (Reapproved 2008).” Farmington Hills, Michigan: American Concrete Institute. 2008b.

ACI 349-06, “Evaluation of Existing Nuclear Safety-Related Concrete Structures.” Farmington Hills, Michigan: American Concrete Institute. 2007.

ACI 318-83, “Building Code Requirements for Reinforced Concrete.” Farmington Hills, Michigan: American Concrete Institute. 1983.

ACI 318-95, “Building Code Requirements for Reinforced Concrete.” Farmington Hills, Michigan: American Concrete Institute. 1995.

ACI 349-85, “Code Requirements for Nuclear Safety Related Concrete Structures.” Farmington Hills, Michigan: American Concrete Institute. 1985.

ACI 349-85, “Code Requirements for Nuclear Safety Related Concrete Structures.” Farmington Hills, Michigan: American Concrete Institute. 1997.

ACI 349.3R, “Evaluation of Existing Nuclear Safety-Related Concrete Structures.” Farmington Hills, Michigan: American Concrete Institute. 2002.

Alhasan, S.J., “Corrosion of Lead,” Vol. 13B, *ASM Handbook*. Materials Park, Ohio: ASM International. pp. 195–204. 2005.

Allen. L.W., and Gary J. Stensland, “Atmospheric Dispersion Study of Deicing Salt Applied to Roads” Part II Final Period July 2002 to June 2004,” January 2006 Issue of Physical Research Report No. FHWA/IL/HRC.2006-1, Illinois Department of Transportation, January 2006 (available from the website http://www.dot.state.il.us/materials/research/pdf/prr149.pdf).

Ambler, H.R., and A.A.J. Bain, “Corrosion of metals in the tropics,” Journal of Applied Chemistry, Vol. 5, pp. 437–467, 1955.

ANSI N14.5-1997, “Standard for Radioactive Materials – Leakage Tests on Packages for Shipments,” New York, NY: American National Standards Institute, 1997.

ANSYS Inc. “FLUENT™ Software Code, Version 14.0,” Canonsburg, PA: ANSYS, Inc. 2013

ASCE/ANSI 11-99, “Guideline for Structural Condition Assessment of Existing Buildings,” American Society of Civil Engineers. 2000.

ASME B&PV Code, Section III, Division 1, Subsection NB, 1983 Edition with Winter 1985 Addenda, New York, NY: American Society of Mechanical Engineers, 1985a.

ASME B&PV Code, Section III, Division 1, Subsection NC, 1983 Edition with Winter 1985 Addenda, New York, NY: American Society of Mechanical Engineers, 1985b.

ASME B&PV Code, Section III, Division 1, Subsection NB, 1998 Edition with 1999 Addenda, New York, NY: American Society of Mechanical Engineers, 1999.

ASME B&PV Code, Section III, Division 1, Subsection NB, 1998 Edition including 2000 Addenda, New York, NY: American Society of Mechanical Engineers, 2000a

ASME B&PV Code, Section III, Division 1, Subsection NC, 1998 Edition including 2000 Addenda, New York, NY: American Society of Mechanical Engineers, 2000b.

ASME B&PV Code, Section III, Division 1, Subsection NB, 2004 Edition with Addenda through 2006, New York, NY: American Society of Mechanical Engineers, 2006.

ASME B&PV Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." The ASME Boiler and Pressure Vessel Code, 2010 edition as approved in 10 CFR 50.55a, New York, NY: American Society of Mechanical Engineers, 2010.

ASME B&PV Code, Section II, "Materials Part D Properties (Customary) "New York, NY: American Society of Mechanical Engineers, 2013.

AREVA TN, "Revision 1 to Transnuclear, Inc. (TN) Application for Amendment 10 to the Standardized NUHOMS® System (Docket No. 72-1004; TAC No. L24052)," Letter E-25506, Columbia MD: AREVA TN, November 7, 2007.

AREVA TN Inc., "Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," Revision 14, Document NUH003.0103, Columbia, MD: AREVA TN, September 2014a.

AREVA TN Inc., "Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," Amendment Number 13 to CoC 1004, Technical Specification for the Standardized NUHOMS® Horizontal Modular Storage System Docket 72-1004, Effective May 24, 2014b.

AREVA TN Inc., "Submittal of Biennial Report of 10 CFR 72.48 Evaluations Performed for the Standardized NUHOMS Systems, CoC 1004, for the Period 07/24/14 to 07/25/16, Docket 72-1004," Letter E-44173, Columbia MD: AREVA TN, July 25, 2016.

Baboian, R. "Galvanic Corrosion", Corrosion: Fundamentals, Testing, and Protection. Vol. 13A. ASM Handbook. ASM International. pp. 210–213. 2003.

Bare, W.C., and L.D. Torgerson, "Dry Cask Storage Characterization Project-Phase 1: CASTOR V/21 Cask Opening and Examination," NUREG/CR-6745, Washington, D.C.: U.S. Nuclear Regulatory Commission, September 2001.

Billone, M.C., T.A. Burtseva, and R.E. Einziger, “Ductile-to-Brittle Transition Temperature for High-Burnup Cladding Alloys Exposed to Simulated Drying-Storage Conditions,” Journal of Nuclear Materials, Volume 433, Issues 1–3, pages 431–448, February 2013.

BISCO Products, Inc., NS-3 Specification Sheet (ADAMS Accession No. ML110730731), June 23, 1986.

Blomqvist G. and M. Gustafsson, “Roadside Salt Exposure under Different Weather Conditions,” Seventh International Symposium on Snow Removal and Ice Control Technology, Transportation Research Circular, E-C126, pp. 159-170, June 2008.

Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation, Material License No. SNM-2505, Docket No. 72-8, “Response to Request for Supplemental Information, RE: Calvert Cliffs Independent Spent Fuel Storage Installation License Renewal Application,” (TAC No- L24475), ADAMS Accession No. ML12212A216, July 27, 2012.

Caskey, G.R., R.S. Ondrejcin, P. Aldred, R.B. Davis, and S.A. Wilson. “Effects of Irradiation on Intergranular Stress Corrosion Cracking of Type 304 Stainless Steel.” Proceedings of 45th NACE Annual Conference, April 23–27, 1990, Las Vegas, Nevada. 1990.

Castaneda, A., F. Corvo, J. J. Howland, and T. Perez, “Atmospheric Corrosion of Reinforced Steel in Tropical Coastal Regions,” Engineering Journal, Vol.17, Issue.2, pp. 1-18, 2013.

Chopra, O., D. Diercks, R. Fabian, Z. Han, and Y. Liu. “Managing Aging Effects on Dry Cask Storage Systems for Extended Long-Term Storage and Transportation of Used Fuel.” FCRD–UFD–2014–000476. ANL–13/15, Rev. 2. Washington, DC: U.S. Department of Energy. 2014.

Cole, I.S., W. D. Ganther, D.A. Paterson, G.A. King, and S.A. Furman, “Holistic Model for Atmospheric Corrosion Part 2 – Experimental Measurement of Deposition of Marine Salts in a Number of Long Range Studies,” Corrosion Engineering, Vol. 38, No. 4, pp. 259-266, 2003.

Corvo, F., N. Betancourt, and A. Mendoza, “The influence of airborne salinity on the atmospheric corrosion of steel,” Corrosion Science, Vol. 37, pp. 1889–1901, 1995.

Croff, A.G., "A User's Manual for the ORIGEN2 Code," ORNL/TM-7175, Oak Ridge National Laboratory, July 1980.

Cumblidge, S.E., M.T. Anderson, S.R. Doctor, “An Assessment of Visual Testing,” NUREG/CR-6860, ML043630040, Washington, DC: U.S. Nuclear Regulatory Commission, November 2004.

S.E. Cumblidge, M.T. Anderson, S.R. Doctor, F.A. Simonen, A.J. Elliott, “A Study of Remote Visual Methods to Detect Cracking in Reactor Components,” NUREG/CR-6943, ML073110060, Washington, DC: U.S. Nuclear Regulatory Commission, November 2007.

Daum, R.S., S. Majumbdar, Y. Liu, and M.C. Billone, “Radial-hydride Embrittlement of High-burnup Zircaloy-4 Cladding,” Journal of Nuclear Science and Technology, Vol. 43, No. 9, p.1054, 2006.

Duke Energy Carolinas, LLC, “Oconee Nuclear Station, Docket No. 72-4, License No. SNM-2503, License Renewal Application for the Site-Specific Independent Spent Fuel Storage Installation (ISFSI) - Response to Requests for Additional Information, License Amendment Request No. 2007-06,” ADAMS Accession No. ML090370066, January 30, 2009.

Einziger, R.E, H. Tsai, M.C. Billone, and B.A. Hilton. NUREG/CR–6831, “Examination of Spent PWR Fuel Rods After 15 Years in Dry Storage.” ANL-03/17. ADAMS Accession No. ML032731021, Washington, DC: U.S. Nuclear Regulatory Commission, September 2003.

EPRI, “PWR Primary Water Chemistry Guidelines: Volume 1, Revision 4,” TR-105714-V1R4. EPRI, Palo Alto, CA: 1999.

EPRI, “BWR Water Chemistry Guidelines - 2000 Revision Non-proprietary version,” BWRVIP-79 TR-103515-R2, Final Report, March 2000.

EPRI, “Climatic Corrosion Considerations for Independent Spent Fuel Storage Installations in Marine Environments.” EPRI 1013524. Palo Alto, California: Electric Power Research Institute. 2006.

EPRI, “Plant Support Engineering: Aging Effects for Structures and Structural Components - Structural Tools,” EPRI Report 1015078, Palo Alto, CA: Electric Power Research Institute, December 2007.

EPRI, “High Burnup Storage Cask Research and Development Project: Final Test Plan,” Palo Alto, CA: Electric Power Research Institute, February 27, 2014.

Farrell, K., “Assessment of Aluminum Structural Materials for Service within the ANS Reflector Vessel,” ORNL/TM-13049, August 1995.

Feliu, S., M. Morcillo, B. Chico, “Effect of distance from sea on atmospheric corrosion rate,” Corrosion, Vol. 55, pp. 883–891, 1999.

Fontana, M.G., “Corrosion Engineering,” New York, New York: McGraw-Hill Book Company, 1986.

Fuhr, K., J. Gorman, J. Broussard, G. White, “Failure Modes and Effects Analysis (FMEA) of Welded Stainless Steel Canisters for Dry Cask Storage Systems,” EPRI-3002000815, Palo Alto, CA: Electric Power Research Institute, 2013.

Gamble, R., “BWRVIP–100–A: BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds.” EPRI–1013396. Palo Alto, CA: Electric Power Research Institute, 2006.

Gavenda, D.J., W.F. Michaud, T.M. Galvin, W.F. Burke, and O.K. Chopra, “Effects of Thermal Aging on Fracture Toughness and Charpy–Impact Strength of Stainless Steel Pipe Welds,” NUREG/CR–6428, ANL–95/47, Washington, DC: NRC, 1996.

Gorman, J., K. Fuhr, J. Broussard, “Literature Review of Environmental Conditions and Chloride-Induced Degradation Relevant to Stainless Steel Canisters in Dry Cask Storage Systems,” EPRI-3002002528, Palo Alto, CA: Electric Power Research Institute, 2014.

Grubb, J.F., T. DeBold, and J.D. Fritz. "Corrosion of Wrought Stainless Steels." ASM Handbook. Vol. 13B. *Corrosion*: Materials. Materials Park, Ohio: ASM International. pp. 54–77. 2005.

Gustafsson, M.E.R., and L.G. Franzen, "Dry Deposition and Concentration of Marine Aerosols in a Coastal Area, SW Sweden," Atmospheric Environment, Vol. 30, No. 6, pp. 977-989, 1996.

H.B. Robinson Steam Electric Plant, Unit No. 2, "Independent Spent Fuel Storage Installation, Docket No. 72-3/License No. SNM-2502, Independent Spent Fuel Storage Installation (ISFSI) Inspection Report," January 31, 1994.

H.B. Robinson Steam Electric Plant, Unit No. 2, "Independent Spent Fuel Storage Installation, Docket No. 72-3/License No. SNM-2502, Independent Spent Fuel Storage Installation (ISFSI) Inspection Report," May 5, 2000.

Hanson, B., H. Alsaed, C. Stockman, D. Enos, R. Meyer, and K. Sorenson. "Used Fuel Disposition Campaign: Gap Analysis to Support Extended Storage of Used Nuclear Fuel, Rev. 0." Richland, Washington: Pacific Northwest National Laboratory. 2012.

Hayashibara, H., M. Mayuzumi, Y. Mizutani, and J-I, Tani, "Effects of temperature and humidity on atmospheric stress corrosion cracking of 304 stainless steel," Corrosion 2008 Conference, Paper No 08492, Houston, TX: NACE International. 2008.

He, X., T.S. Mintz, R. Pabalan, L. Miller, G. Oberson, "Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Atmospheric Chloride and Non-Chloride Salts," NUREG/CR-7170, ML14051A417, Washington, DC:U.S. Nuclear Regulatory Commission, 2014.

Hilsdorf, H.K., J. Kropp, and H.J. Koch, "The Effects of Nuclear Radiation on the Mechanicals Properties of Concrete", *American Concrete Institute Special Publication SP-55*, Farmington Hills, MI: American Concrete Institute, pp. 223-251, 1978

Hossain, K.M.A. and S.M. Easa, "Spatial distribution of marine salts in coastal region using wet candle sensors" IJRRAS, vol.7, No.3, pp.228-235, June 2011.

Johnson, A.B., Jr. and E. R. Gilbert, "Technical Basis for Storage of Zircaloy Clad Spent Fuel in Inert Gases," PNL-4835, U.S. Department of Energy, Richland, WA: Pacific Northwest Laboratory, 1983.

Kain, R. "Marine Atmospheric Stress Corrosion Cracking of Austenitic Stainless Steel." *Materials Performance*. Vol. 29, No. 12. pp. 60–62. 1990.

Kamimura, K., "Integrity Criteria of Spent Fuel for Dry Storage in Japan" Proc. International Seminar on Interim Storage of Spent Fuel, ISSF 2010, Tokyo Nov 2010.

Levy, I.S., C.E. Beyer, B.A. Chin, E.R. Gilbert, E.P. Simonen, and A.B. Johnson, Jr, "Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas," PNL-6189 (UC-85) U.S. Department of Energy, Richland, WA: Pacific Northwest Laboratory, May 1987.

Magee, J.H. "Wear of Stainless Steels", Friction, Lubrication, and Wear Technology. ASM Handbook, Vol 18. ASM International. pp. 710–724. 1992.

McConnell Jr., J.W., A.L. Ayers, Jr., M.J. Tyacke, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," NUREG/CR-6407, INEL-95/0551, Washington, D.C.: U.S. Nuclear Regulatory Commission, February 1996.

Meira, G.R., C. Anderson, C. Alonso, I.J. Padaratz, and J.C. Borba, "Modeling seasalt transport deposition in marine atmosphere zone – A tool for corrosion studies," Corrosion Science, Vol. 50, pp. 2724-2731, 2008.

Meroney, R.N., "CFD Prediction of Cooling Tower Drift," Civil Engineering Department, Colorado State University, Fort Collins, Colorado, August 5, 2005.

Meyer, R.M., A.F. Pardini, J.M. Cuta, H.E. Adkins, A.M. Casella, A. Qiao, A.A. Diaz, and S.R. Doctor. "NDE to Manage Atmospheric SCC in Canisters for Dry Storage of Spent Fuel: An Assessment." PNNL-22495. Richland, Washington: Pacific Northwest National Laboratory. 2013.

Naus, D.J., "Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures - A Review of Pertinent Factors," NUREG/CR-6927, Washington D.C.: U.S. Nuclear Regulatory Commission, February 2007.

NEI, NEI 14-03, "Guidance for Operations-Based Aging Management for Dry Cask Storage," Revision 0, Nuclear Energy Institute, September 2014.

NEI, NEI 14-03, "Format, Content, and Implementation Guidance for Cask Storage Operations-Based Aging Management," Revision 1, September 2015.

Nikolaev, Yu., A.V. Nikolaeva, and Ya.I. Shtrombakh. "Radiation Embrittlement of Low-Alloy Steels." *International Journal of Pressure Vessels and Piping*. Vol. 79. pp. 619–636. 2002.

NRC, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," NUREG-0612, Washington, D.C.: U.S. Nuclear Regulatory Commission, 1980.

NRC, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, Revision 1. Washington, D.C.: U.S. Nuclear Regulatory Commission, 1983.

NRC, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Interim Staff Guidance, SFST-ISG-11, Rev. 2, Washington, D.C.: U.S. Nuclear Regulatory Commission, June 30, 2002.

NRC, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Interim Staff Guidance, SFST-ISG-11, Rev. 3, Washington D.C.: U.S. Nuclear Regulatory Commission, November 17, 2003.

NRC, "Surry Independent Spent Fuel Storage Installation," License No. SNM-2501 License Renewal Safety Evaluation Report Docket No. 72-2, ADAMS Accession No. ML050590266, February 25, 2005a.

NRC, “H. B. Robinson Independent Spent Fuel Storage Installation,” License No. SNM-2502 License Renewal Safety Evaluation Report Docket No. 72-3, ADAMS Accession No. ML050890397, March 30, 2005b.

NRC, “Meteorological Monitoring Programs for Nuclear Power Plants,” Regulatory Guide 1.23, Rev. 1, ADAMS Accession No. ML070350028, Washington, D.C.: U.S. Nuclear Regulatory Commission March 2007.

NRC, “Oconee Nuclear Station Independent Spent Fuel Storage Installation,” License No. SNM-2503 License Renewal, Safety Evaluation Report, Docket No. 72-04, ADAMS Accession No. ML091520159, May 29, 2009.

NRC, “Standard Review Plan for Dry Cask Storage Systems at a General License Facility,” NUREG-1536 Revision 1, Washington D.C.: U.S. Nuclear Regulatory Commission, July 2010a.

NRC “Fuel Retrievability,” Interim Staff Guidance, SFST-ISG-2, Rev 1, Washington, D.C.: U.S. Nuclear Regulatory Commission, February 2010b.

NRC, “Pressure and Helium Leakage Testing of the Confinement Boundary of Spent Fuel Dry Storage Systems,” ISG-25, ADAMS Accession No. ML101970493, Washington, DC, 2010c.

NRC, “Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Washington, D.C.: U.S. Nuclear Regulatory Commission, 2010d.

NRC, “Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance,” NUREG-1927, ADAMS Accession No. ML111020115, Washington, D.C.: U.S. Nuclear Regulatory Commission, March 2011a.

NRC, “Ongoing Review of Operating Experience,” Final License Renewal Interim Staff Guidance LR-ISG-2011-05 (ADAMS Accession No. ML12044A215) Washington, D.C.: U.S. Nuclear Regulatory Commission, 2011b
.
NRC “Potential Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canister,” Information Notice 2012-20, Washington D.C.: U.S. Nuclear Regulatory Commission, November 14, 2012.

NRC, “Identification and Prioritization of the Technical Information Needs Affecting Potential Regulation of Extended Storage and Transportation of Spent Nuclear Fuel.” Washington, DC: U.S. Nuclear Regulatory Commission. May 2014.

NRC, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel,” NUREG-1927, Revision 1, ADAMS Accession No. ML111020115. Washington, D.C.: U.S. Nuclear Regulatory Commission, June 2016.

NWTRB. “Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel.” Washington, DC: Nuclear Waste Technical Review Board. 2010.

Odette, G.R. and G.E. Lucas. “Embrittlement of Nuclear Reactor Pressure Vessels.” *Journal of Metals*. Vol. 53, Issue 7. pp. 18-22. 2001.

ORNL, “MCNP/MCNPX – Monte Carlo N-Particle Transport Code System Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries,” CCC-730, Oak Ridge National Laboratory, RSICC Computer Code Collection, January 2006.

Revie, R.W., *Uhlig’s Corrosion Handbook*, Second Edition, Hoboken, New Jersey, John Wiley and Sons. 2000.

H.B. Robinson Steam Electric Plant, Unit No. 2, “Independent Spent Fuel Storage Installation, Docket No. 72-3/License No. SNM-2502, Independent Spent Fuel Storage Installation (ISFSI) Inspection Report,” January 31, 1994.

H.B. Robinson Steam Electric Plant, Unit No. 2, “Independent Spent Fuel Storage Installation, Docket No. 72-3/License No. SNM-2502, Independent Spent Fuel Storage Installation (ISFSI) Inspection Report,” May 5, 2000.

Roffman A., and Lowell D. Vleck, “The state of the art of measuring and predicting cooling tower drift and its deposition,” Journal of the Air Pollution Control Association, Vol.24, No.9, pp. 855-859, 1974.

Scaglione, J.M., G. Radulescu, W. J. Marshall, K. R. Robb. “A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages,” NUREG/CR-7203, Oak Ridge National Laboratory, ORNL/TM-2013/92, Oak Ridge, TN, 2015

Shirai, K., J. Tani, H. Takeda, M. Wataru, and T. Saegusa, “SCC Evaluation Test of Multi-Purpose Canister,” Proceedings of 2011 Water Reactor Fuel Performance Meeting, Paper No. T4-012, Chengdu, China, September 11-14, 2011a.

Shirai, K., J. Tani, and T. Saegusa, “Study on Interim Storage of Spent Nuclear Fuel by Concrete Cask for Practical Use – Feasibility Study on Prevention of Chloride Induced Stress Corrosion Cracking for Type 304L Stainless Steel Canister,” CRIEPI Report No. N10035, Tokyo, Japan, 2011b.

Shirai, K., J. Tani, H. Takeda, M. Wataru, H. Takeda and T. Saegusa, “SCC Evaluation of Multi-Purpose Canister,” International Conference on High Level Radioactive Waste Management, Albuquerque, New Mexico, Paper No. 288, 2011c.

Sindelar, R.L., A.J. Duncan, M.E. Dupont, P.-S. Lam, M.R. Louthan, Jr., and T.E. Skidmore. NUREG/CR–7116, “Materials Aging Issues and Aging Management for Extended Storage and Transportation of Spent Nuclear Fuel.” Washington, DC: U.S. Nuclear Regulatory Commission. 2011.

Subacz, J., “LNP – Cooling Tower Plume Deposition Analysis,” CH2MHILL, Document No: 338884-PI-03-14, USNRC ADMAS ML12178A143, May 22, 2008.

Szklarska-Smialowska, Z., “Pitting Corrosion of Metals,” Houston, TX: National Association of Corrosion Engineers, 1986.

Tanaka, Y., H. Kawano, H. Watanabe, and T. Nakajo, “Study on Cover Depth for Prestressed Concrete Bridges in Airborne-Chloride Environment.” PCI Journal, pp.42-53, 2006.

Waldrop, K., W. Bracey, K. Morris, C. Bryan, D. Enos, “Calvert Cliffs Stainless Steel Dry Storage Canister Inspection,” EPRI-1025209, Palo Alto, CA: Electric Power Research Institute, 2014.

Wang, J-A. and H. Wang. NUREG/CR-7198, “Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications.” ADAMS Accession No. ML15139A389. Washington, DC: U.S. Nuclear Regulatory Commission. May 2015.

Was, G.S., J. Busby, and P.L. Andresen. “Effect of Irradiation on Stress-Corrosion Cracking and Corrosion in Light Water Reactors.” *Metalworking: Bulk Forming.* Vol. 13C. *ASM Handbook*. ASM International. pp. 386–414. 2006.