Attachment 1

Request for Additional Information Responses

NAC-MPC CASK SYSTEM, Revision 22A

NAC INTERNATIONAL

NON-PROPRIETARY RESPONSES TO THE

UNITED STATES
NUCLEAR REGULATORY COMMISSION

REQUEST FOR ADDITIONAL INFORMATION

December 21, 2021

FOR REVIEW OF THE NAC-MPC CASK SYSTEM

(Docket No. 72-1025)

NAC International

March 2022

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Appendix A - Aging Management Programs
RAI A-2:

[Applies to NAC-UMS and NAC-MPC]
Justify how the proposed inspection methodology for the supplemental examination in the Localized Corrosion and Stress Corrosion Cracking of Welded Stainless-Steel TSC AMP will be capable of identifying and sizing a crack.

The Localized Corrosion and Stress Corrosion Cracking of Welded Stainless-Steel TSC AMP includes a supplemental examination to further examine major indications of corrosion. In the Acceptance Criteria program element, item 6.4, the proposed examination methodology is "VT-3, VT-1, or other interrogative nondestructive techniques."

It is unclear to staff how the proposed inspection methodology will be capable of identifying and sizing cracking, as the listed techniques are not generally considered to be appropriate for that task. Section 3.4.3.1 of the renewal application states that the subject AMP intends to follow the guidelines in EPRI Report TR-3002008193 (EPRI, 2017); however, the EPRI guidelines include the use of surface or volumetric techniques when examining major indications of corrosion at or near a weld. Similarly, ASME Code Case N-860 (ASME, 2020) requires the use of surface or volumetric techniques.

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

References

ASME. American Society of Mechanical Engineers Boiler and Pressure Vessel Code Case N-860, "Inspection Requirements and Evaluation Standards for Spent Nuclear Fuel Storage and Transportation Containment Systems," July 2020.

EPRI. Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless Steel Canisters, Technical Report No. 3002008193, Electric Power Research Institute, 2017.

NAC International Response to RAI A-2:

NAC has revised AMP 1 Localized Corrosion and Stress Corrosion Cracking to incorporate the requirements from Section -2400 of ASME Code Case N-860 into Section 6.4 for supplemental examinations using surface or volumetric examination techniques per IWA-2220 or IWA-2230, respectively, or the equivalent. An analysis may be performed per Section -2440 of the Code Case if a supplemental examination is not possible or is unable to provide sufficient data. Other Elements have been revised as detailed below.

Summary of Changes:

Revised AMP-1 Localized Corrosion and Stress Corrosion Cracking of Welded Stainless Steel Transportable Storage Canisters Section Element 4 has been revised to read as follows:

Method or Technique

Aging effects are detected and characterized by:

- General visual examination using direct or remote methods of the TSC accessible external surfaces away from the weld region for localized corrosion and anomalies.
- Visual screening examination by direct or remote means of accessible TSC welds, associated HAZs, and known areas of removed temporary attachments and weld repairs using qualified VT-3 methods and equipment to identify corrosion products that may be indicators of localized corrosion and SCC.
- An assessment examination meeting the requirements of VT-1 is required if the screening examination identifies any visual anomaly that is not consistent with prior results or is identified for the first time.
- A supplemental examination is required for any visual anomaly within the weld region that is classified as a major indication as discussed in Section 6, Acceptance Criteria.
- The extent of coverage shall be maximized subject to the limits of accessibility.

Revised AMP-1 Localized Corrosion and Stress Corrosion Cracking of Welded Stainless Steel Transportable Storage Canisters Section Element 6.2 has been revised to read as follows:

6.2. Acceptance Criteria for TSC Welds and HAZ Areas Using VT-3:

a. If no visual indications of corrosion or SCC are present (i.e. visually clean) no additional action is required.
b. An assessment examination meeting the requirements of VT-1 is required if the screening examination identifies any visual anomaly that is not consistent with prior results or is identified for the first time.
c. If a corrosion indication meets any of the following, it should be considered a major indication and subject to supplemental examinations per 6.4:
    - Cracking of any size
    - Corrosion products having a linear or branching appearance
    - Evidence of pitting corrosion, under deposit corrosion, or etching with measurable depth (removal/attack of material by corrosion.)”

Revised AMP-1 Localized Corrosion and Stress Corrosion Cracking of Welded Stainless Steel Transportable Storage Canisters Section Element 6.4 has been revised to read as follows:

6.4 A supplemental examination of major indications shall be performed in accordance with Section-2400 of ASME Code Case N-860 as detailed below:

a. If a surface technique is used to size a flaw, the examination shall be performed in accordance with IWA-2220 or equivalent
b. If a volumetric technique is used to size a flaw, the examination shall be performed in accordance with IWA-2230 or equivalent
c. If a supplemental examination is not possible or unable to provide sufficient data, an analysis shall be used when justified in accordance with N-860 Section -2440
d. The required actions of N-860 Section -2432 shall be followed, depending on the results of the supplemental examinations or N-860 Section 2441 if analysis is employed.

In addition, Section 3.4.3.1 has been revised as follows:

"3.4.3.1 AMP-1 Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSCs)

In lieu of utilizing the proposed inspection guidelines and acceptance criteria proposed in Table 6.2 of NUREG-2214, NAC-MPC users intend to utilize the inspections guidelines and acceptance criteria provided in EPRI Report TR-3002008193 [3.9.16] and ASME Code Case N-860 (3.9.368) for supplemental examination of major indications as documented in the proposed AMP. To support identification of the most susceptible TSCs to SCC, all NAC-MPC user ISFSIs and loaded TSCs were evaluated and ranked utilizing EPRI Report TR-3002005371 [3.9.15].

For examination of TSC welds and heat affected zones (HAZs) qualified VT-3 inspection methods will be utilized. Certain inspection results require a supplemental examination per ASME Code Case N-860 (3.9.368). If a supplemental examination is not possible or is unable to provide sufficient data, an analysis shall be performed as provided for by Section-2440. TSC surfaces outside of the welds and HAZs, a direct or remote general visual inspection will be conducted. If issues are identified during the general visual inspection of non-welded TSC surfaces, supplemental examinations can be performed with VT-3 and VT-1 equipment and methods."

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Appendix A - Aging Management Programs
RAI A-3:

[Applies to NAC-UMS and NAC-MPC]

In the AMR Tables and the proposed revision to FSAR Chapter 14, clarify if the Internal and External [external not applicable to the MPC] VCC Metallic Components Monitoring AMPs are activities credited to manage the effects of aging.

FSAR Tables 14.3-6 [14.3-5 for the MPC] and 14.3-7 in Appendix C of the renewal application include proposed AMPs for the inspection of metallic VCC components. However, there are no AMR line items that credit the use of these AMPs to manage the effects of aging. The renewal application states that aging of the metallic VCC components is addressed by the TLAA that concluded that corrosion will not prevent the VCCs from fulfilling their important-to-safety functions.

It is unclear to the staff if the Internal VCC Metallic Components Monitoring AMP and External VCC Metallic Components Monitoring AMP [external not applicable to the MPC] are relied on to manage the effects of aging. Provide clarifications to the AMR tables and proposed FSAR revisions to establish the purpose of the subject AMPs and whether they are to be performed by general licensees to fulfill aging management requirements of the renewed CoC.

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

NAC International Response to RAI A-3:

The Chapter 14 Tables 14.3-2 and 14.3-3 have been revised to identify Galvanic Corrosion as an aging mechanism that is managed by the external VCC metallic components AMP. Additionally, the corresponding tables in the application Tables 3.2-2 and 3.2-3 have been revised to incorporate Galvanic Corrosion as an applicable aging mechanism that is managed by the appropriate AMP.

It is the intent of the Internal and External VCC Metallic Components Monitoring AMPs to be performed by all General Licenses on the schedules defined in the AMPs to inspect for any loss of components, excessive or unusual coating damage, excessive general corrosion and galvanic corrosion of the identified components. The TLAA is used to address normal coating wear and tear on both inner and outer metallic surfaces such that there is flexibility available to the licensee for external coating repair activities. Remote internal coating repairs are not anticipated while bounded by the TLAA.

Summary of Changes:

Table 14.3-2 and 14.3-3, and Tables 3.2-2 and 3.2-3 were also revised to reference the requirement to inspect NAC-MPC internal and external components in accordance with the referenced AMPs in addition to taking credit for the applicable TLAAs. The AMP inspections will be performed as required by the AMP frequency.

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Appendix A - Aging Management Programs
RAI A-6:

[Applies to NAC-UMS and NAC-MPC]

State how visual inspection parameters will be controlled to ensure that the AMPs for the VCC and transfer cask components will be capable of identifying degradation.
The AMPs detailed in Tables A-2, A-3, and A-6 [A-5 rather than A-6 for the MPC] of the renewal application for the VCC and transfer cask rely on general visual inspections to identify degradation. The AMPs do not cite consensus code criteria nor alternative approaches that describe how procedures are controlled to ensure that inspectors will use sufficient resolution and lighting to identify the parameters monitored. Revise the AMPs for the VCC and transfer cask, as appropriate, to clarify the expectations of the general licensees for controlling visual inspection parameters.
This information is required to demonstrate compliance with 10 CFR 72.240(c).

NAC International Response to RAI A-6:

AMP-2, Internal VCC Metallic Components Monitoring. AMP-3, External VCC Metallic Components Monitoring, and AMP-5, Transfer Casks and Transfer Adapters have been revised to reference ASME Code, Section XI, Division 1, Subsection IWE, 2007 for examination requirements including IWE-2310 Visual Examination, IWE-2311 General Visual Examination, IWE-2330 Personnel Qualification, and IWE-3511 General Visual Examination of Coated and Uncoated Areas Acceptance Standards. Note: The Transfer Cask AMP is AMP-5 in both the NAC-UMS and NAC-MPC applications.

Summary of Changes:

AMP-2, AMP-3 and AMP-5 have revised to add the following to the AMP’s Elements as follows:

AMP-2 Element 4 has been revised to read as follows:

Method or Technique

Aging effects are detected and characterized by:

- General visual examination using direct or remote methods of the accessible VCC internal metallic components for corrosion resulting in significant loss of metal, component displacement or degradation, or air passage blockage.

- The extent of inspection coverage shall be maximized, subject to the limits of accessibility.
- Visual examinations shall comply with IWE-2311 requirements or their equivalent.
- Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent.

AMP-2 Element 6 has been revised to read as follows:

The acceptance criteria for the visual inspections are:

- No obvious loss of base metal.
- No indication of displaced or degraded components.
- No indications of damaged bolts or bolt holes (in cases where VCC lid is removed).
- The inspected condition of the examined area is acceptable per IWE-3511 standard or their equivalent.

AMP-3 Element 3 has been revised to read as follows:

Parameters to be inspected and/or monitored on external VCC coated steel surfaces will include:

- Visual evidence of significant coating loss or galvanic corrosion which left uncorrected could result in obvious loss of base metal.
- Visual evidence of loose or missing bolts, galvanic corrosion, physical displacement, and other conditions indicative of loss of preload on VCC lid and lifting lug bolting, as applicable.

AMP-3, Element 4 has been revised to read as follows:

<u>Method or Technique</u>

Aging effects are detected and characterized by:

- General visual examination using direct methods of the external VCC metallic components for significant corrosion or significant coating loss resulting in loss of base metal.
- The extent of inspection shall cover all normally accessible VCC lid surfaces, VCC lid flange, exposed steel surfaces of the inlet and outlet vents, VCC lifting lugs, and VCC lid and lift lug bolting.
- Visual examinations shall comply with IWE-2311 requirements or their equivalent.
- Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent.

AMP-3, Element 6 has been revised to read as follows:

The acceptance criteria for the visual inspections are:

- No active corrosion resulting in obvious, loss of base metal.

- Areas of coating failures must remain bounded by the corrosion analysis of TLAA 30013-2002, latest revision, or are entered into the corrective action program.
- No indications of loose bolts or hardware, displaced parts.
- The inspected condition of the examined area is acceptable per the IWE-3511 standard or their equivalent.

AMP-5 has been revised as follows:

AMP-5 Element 4 has been revised to read as follows:

Method or Technique

Aging effects are detected and characterized by:

- General visual examinations using direct methods of the TFR/Transfer Adapter steel surfaces for cracking, corrosion or wear resulting in loss of base metal or coating damage which left uncorrected could result in loss of base metal.
- The extent of inspection coverage will include all normally accessible and visible TFR/Transfer Adapter interior cavity and exterior surfaces. Also inspected are the retaining ring and associated bolting, shield doors and shield door rails.
- Dye penetrant (PT) examinations of accessible trunnion surfaces for the presence of fatigue cracks in accordance with ASME Code, Section III, Subsection NF, NF-5350.
- Visual examinations shall comply with IWE-2311 requirements or their equivalent.
- Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent.

AMP-5 Element 6 has been revised to read as follows:

For accessible surfaces, including trunnions, acceptance criteria are:

- No obvious, loss of material from the base metal.
- No large areas of coating failures which could expose base metal to active corrosion
- No areas of wear resulting in obvious loss of base metal.
- Successful completion of dye penetrant (PT) examinations of accessible trunnion surfaces for the presence of fatigue cracks in accordance with ASME Code, Section III, Subsection NF, NF-5350.
- The inspected condition of the examined area is acceptable per the acceptance standards of IWE-3511 or their equivalent.

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Appendix A - Aging Management Programs
RAI A-7:

[Applies to NAC-UMS and NAC-MPC]

Clarify the proposed changes to FSAR Sections 9.2.1 and 9.2.2 [9.A.3.1 and 9.A.3.2 for the MPC-LACBWR] with respect to when the annual inspections of the VCCs and transfer casks will be replaced by the associated AMP activities for the individual storage systems.

The proposed FSAR Sections 9.2.1 and 9.2.2 [9.A.3.1 and 9.A.3.2 for the MPC] in Appendix C of the renewal application both state:

> After the approval of the 40-year CoC renewal term General Licenses will adopt the aging management programs (AMPs) as described in Chapter 14 for *their sites period of extended operation (PEO)*[emphasis added].

The staff notes that the emphasized text above does not accurately describe the renewed licensing basis, as general licensee sites do not have a PEO. Rather, the PEO applies to the CoC, and by extension, to each individual dry storage system (as described in NUREG-1927, Section 3.6.2, "Commencement of AMP(s) for CoC Renewals," and Appendix F, "Storage Terms" (NRC, 2016)). Unless otherwise specified in the CoC or FSAR, AMPs are considered to apply to each individual dry storage system when that storage system enters its renewed storage period. As a result, provide clarity to the FSAR with respect when AMP activities begin for the individual storage systems.

This information is required to demonstrate compliance with 10 CFR 72.240(c).

Reference

NRC. NUREG–1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel." Revision 1. ADAMS Accession No. ML16179A148. Washington, DC: U.S. Nuclear Regulatory Commission. 2016.

NAC International Response to RAI A-7:

The subject sections of the NAC-MPC FSAR Chapter 9 (9.2) and Chapter 9A (9.A.3.1 and 9.A.3.2) have been revised to correct the proper start of AMP implantation in accordance with NUREG-1927 requirements.

Summary of Changes:

FSAR Section 9.2 has been revised to read as follows:

An AMP for a renewed CoC commences at the end of the initial storage period for each loaded NAC-MPC System. Once the AMP has been implemented for the renewed CoC on a cask system, the performance of the AMP will replace specified maintenance inspections as detailed in Section 9.2.

FSAR Section 9.A.3.1 has been revised to read as follows:

An AMP for a renewed CoC commences at the end of the initial storage period for each loaded NAC-MPC System. Once the AMP has been implemented for the renewed CoC on a cask system, the performance of the AMP will replace specified maintenance inspections as detailed in Section 9.A.3.1.

FSAR Section 9.A.3.2 has been revised to read as follows:

An AMP for a renewed CoC commences at the end of the initial storage period for each loaded NAC-MPC System. Once the AMP has been implemented for the renewed CoC on a cask system, the performance of the AMP will replace specified maintenance inspections as detailed in Section 9.A.3.2.

Enclosure 1

APPLICATION FOR RENEWAL OF THE NAC-MPC SYSTEM CoC

Enclosure 1

APPLICATION FOR RENEWAL OF THE NAC-MPC SYSTEM CoC

ENCLOSURE 1

APPLICATION FOR RENEWAL OF THE NAC-MPC SYSTEM CoC

CONTENTS

APPENDIX

LIST OF TABLES

ACRONYMS AND ABBREVIATIONS

ACI	American Concrete Institute
ALARA	As Low As Reasonably Achievable
AMA	Aging Management Activity
AMP	Aging Management Program
AMR	Aging Management Review
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASR	Akali Silica Reaction
ASTM	American Society of Testing and Materials
BWR	Boiling Water Reactor
CLB	Current Licensing Basis
CFR	Code of Federal Regulations
CH	Certificate of Compliance Holder
CISCC	Chloride Induced Stress Corrosion Cracking
cm	centimeter
CoC	Certificate of Compliance
CR	Subcriticality
CY	Connecticut Yankee
DEF	Delayed Ettringite Formation
DFC	Damaged Fuel Can
DFSM	Division of Spent Fuel Management
DHC	Delayed Hydride Cracking
DOE	U.S. Department of Energy
DPC	Dairyland Power Cooperative
E-C	Embedded (Concrete) Environment
EPRI	Electric Power Research Institute
FB	Fuel Basket
FE	Fully Encased
FOC	Fuel Only Can (DFC)
ft	Foot/Feet
FSAR	Final Safety Analysis Report
GL	General Licensees
GWd/MTU	Gigawatt-Days per Metric Tonne Uranium
HAZ	Heat Affected Zone
HBU	High Burnup
IFA	Irradiated Fuel Assembly
IFBA	Integral Fuel Burnable Absorber
in	Inch/Inches
ISFSI	Independent Spent Fuel Storage Installation
ITS	Important to Safety
kW	kilowatt
LACBWR	La Crosse Boiling Water Reactor
lbs	Pounds
MeV	Million Electron Volts
MIC	Microbial Induced Corrosion
MPC	Multi-Purpose Canister

MWd/MTU	Megawatt-Days per Metric Tonne Uranium
NAC	NAC International, Inc.
N/A	Not Applicable
NDE	Nondestructive Examination
NFPA	National Fire Protection Association
NITS	Not Important to Safety
NMSS	NRC Office of Nuclear Material Safety and Safeguards
NOAA	National Oceanic and Atmospheric Administration
NQ	Non-Quality
NRC	Nuclear Regulatory Commission
OD	Air-Outdoor Environment
OE	Operating Experience
PEO	Period of Extended Operation
ppm	parts per million
PT	Dye Penetrant Examination
PWR	Pressurized Water Reactor
RE	Retrievability
RT	Radiographic Examination
RCA	Radiation Control Area
SAR	Safety Analysis Report
SCC	Stress Corrosion Cracking
SD	Shield Door
SER	Safety Evaluation Report
SFA	Spent Fuel Assembly
SFP	Spent Fuel Pool
SFPO	Spent Fuel Project Office
SH	Sheltered Environment
SNF	Spent Nuclear Fuel
SSC	Structure, System and Component
SR	Structural Integrity
STC	Storable Transport Cask
TFR	Transfer Cask
TH	Thermal/Heat Removal
TLAA	Time Limited Aging Analysis
TMI	Three Mile Island
TS	Technical Specification
TSC	Transportable Storage Canister
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Examination
VCC	Vertical Concrete Cask
VT	Visual Examination
YR	Yankee Rowe
YAEC	Yankee Atomic Electricity Company

1.0 GENERAL INFORMATION

The NAC International Multi-Purpose Canister Storage System (hereafter referred to as the NAC-MPC System) is approved under 10 CFR 72, Subpart K (Docket No. 72-1025) 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. The NAC-MPC System Certificate of Compliance (CoC) was initially issued on April 10, 2000 with an expiration date of April 10. 2020. NAC International (NAC), as the Certificate Holder (CH) of the NAC-MPC System CoC No. 1025 [1.3.1.a through 1.3.1.i], is applying for renewal of CoC No.1025 for a term of 40 years in accordance with 10 CPR 72.240(a).

Additionally, NAC is applying for renewal of the initial NAC-MPC System CoC and Amendments 1 through 8.

The requested 40-year CoC renewal term will extend the CoC expiration date to April 10, 2060. The NAC-MPC System CoC renewal application includes information required by 10 CFR 72.240(c), including:

(1) The design basis information as documented in the most recent updated Final Safety Analysis Report (FSAR) [1.3.2.m.] as required by 10 CFR 72.248;
(2) Time-Limited Aging Analyses (TLAAs) that demonstrate that Structures, Systems, and Components (SSC) Important-to-Safety (ITS) will continue to perform their intended function for the requested period of extended operation; and
(3) A description of the Aging Management Programs (AMPs) for management of issues associated with aging that could adversely affect Structures, Systems, and Components (SSCs) Important to Safety (ITS).

In accordance with 10 CFR 72.240(d), the NAC-MPC System CoC renewal application demonstrates that the storage of SNF has not, in a significant manner, adversely affected structures, systems, and components important to safety.

1.1 BACKGROUND INFORMATION

1.1.1 NAC-MPC CoC and Amendment History

The initial NAC-MPC System CoC [1.3.1.a] was issued on April 10, 2000 based on NAC-MPC Safety Analysis Report (SAR), Revision 5. The original CoC approved the NAC-MPC System design for the Yankee Atomic Electric Company's (YAEC) Yankee Rowe Nuclear Station designated the Yankee-MPC (YR-MPC) system. The system included a Transportable Storage Canister (TSC) provided with a fuel basket designed to accommodate up to thirty-six (36) Yankee-class PWR fuel assemblies; a vertical concrete cask (VCC); and a Transfer Cask (TFR) sized to accommodate the YR-MPC TSC.

Subsequently, eight (8) amendments were issued to the NAC-MPC System CoC. A summary of the NAC-MPC System CoC amendment history is provided in the following paragraphs, including a general description of the changes and reasons for each amendment.

- **Amendment No. 1**: By application dated September 29, 2000, as supplemented October 5, 2000, March 16, April 6 and July 27, 2001, NAC requested NRC approval of an amendment to CoC No. 1025 for the NAC-MPC System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. The proposed amendment requested: (1) an alternate Yankee-MPC fuel basket design with enlarged fuel tubes in the corner locations; (2) an increase in operational time limits for canister loading, closure and transfer provided in the Technical Specifications to allow for canister heat loads that are lower than the design basis heat load; (3) revisions to the Technical Specifications for canister surface contamination to maintain doses to workers As Low As Reasonably Achievable (ALARA); and (4) minor revisions to some of the drawings to reflect changes identified during cask and component fabrication. The request, as supplemented, was approved by the NRC in Amendment No. 1 [1.3.1.b] and was effective November 13, 2001.

- **Amendment No. 2**: By application dated May 19, 2000, as supplemented September 6, October 2 and 12, 2000, and April 13, September 6, October 5, 10 and 15, and November 21, 2001, NAC requested NRC approval of an amendment to CoC No. 1025 for the NAC-MPC System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. The original CoC, as amended, authorized the storage of up to 36 fuel assemblies from the Yankee Rowe (YR) pressurized water reactor (PWR). The proposed amendment requested NRC approval to store the spent nuclear fuel from the decommissioned Connecticut Yankee (CY) Haddam Neck power plant in the NAC-MPC System. The CY-MPC system changes included: (1) increasing the length of the TSC, VCC and Transfer Cask to accommodate the longer CY fuel; (2) a new fuel basket designed for up to 26 CY fuel assemblies with an alternate 24 fuel assembly configuration; and (3) Transfer Cask shielding and length increased to accommodate the CY fuel. Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features) of the certificate were revised in their entirety following the standard technical specification format in NUREG-1745, "Standard Format and

Content for Technical Specifications for 10 CFR 72 Cask Certificates of Compliance." Furthermore, the certificate format was revised to make the conditions more accurate and eliminate duplication. The request, as supplemented, was approved by the NRC in Amendment No. 2 [1.3.1.c] and was effective May 29, 2002.

- **Amendment No. 3**: By applications dated April 18, 2002, May 15, 2002, and January 17, 2003, as supplemented on July 17, 2002, and October 3, 2002, NAC requested NRC approval of an amendment to CoC No. 1025 for the NAC-MPC System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. NAC requested changes to the Certificate of Compliance (CoC), including its attachments, and revision to the Final Safety Analysis Report (FSAR). The proposed amendment requested: (1) incorporation of fuel enrichment fabrication tolerances into the Yankee Class fuel parameters; (2) incorporation of fuel assemblies with up to 20 damaged fuel rods, recaged fuel assemblies, the Yankee Rowe damaged fuel can (DFC), and YR fuel assembly weights up to 950 pounds; (3) revision to the average surface dose rate limits for the concrete cask; (4) incorporation of administrative changes to the ASME Code Alternatives for the NAC-MPC canister; (5) corrections to the Connecticut Yankee (CY) maximum fuel enrichment, maximum initial uranium mass, and maximum burnup parameters; and (6) incorporation of editorial and administrative changes. The request, as supplemented, was approved by NRC in Amendment No. 3 [1.3.1.d] and was effective October 1, 2003.

- **Amendment No. 4**: By application dated August 1, 2003, as supplemented on September 5, and November 3, 2003, NAC requested NRC approval of an amendment to CoC No. 1025 for the NAC-MPC System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. NAC requested changes to the CoC, including its attachments, and revision to the Final Safety Analysis Report (FSAR). The requested changes were to: (1) increase vacuum drying time limits; (2) increase canister in transfer cask time limits; (3) revise fuel cooldown requirements; (4) delete canister removal from concrete cask requirements; (5) revise surface contamination removal time limits; and (6) revise allowable contents fuel assembly limits. The request, as supplemented, was approved by the NRC in Amendment No. 4 [1.3.1.e] and was effective October 27, 2004.

- **Amendment No. 5**: By application dated July 17, 2006, and supplement dated September 13, 2006, NAC requested NRC approval of an amendment to CoC No. 1025 for the NAC-MPC System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. NAC requested NRC to amend CoC No. 1025 for the NAC-MPC System to revise technical specifications (TS) to incorporate changes to the reporting and monitoring requirements, and incorporate guidance from NRC Interim Staff Guidance, ISG-22, "Potential Rod Splitting Due to Exposure to Oxidizing Atmosphere During Short-Term Cask Loading Operations in LWR or Other Uranium Oxide Fuel." NAC also requested in its supplement to the amendment request that the CoC be updated to remove the requirement for installation of tamper-indicating devices on the

VCC and to make this requirement optional. The request, as supplemented, was approved by the NRC in Amendment No. 5 [1.3.1.f] and was effective July 24, 2007.

- **Amendment No. 6**: By application dated January 16, 2009, as supplemented February 11, 2009, April 1, 2009, April 30, 2009, September 22, 2009, and January 8, 2010, NAC requested NRC approval of an amendment to CoC No. 1025 for the NAC-MPC System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. NAC requested approval to store, in its NAC-MPC System spent fuel assemblies from the decommissioned Dairyland Power Cooperative (DPC) LaCrosse Boiling Water Reactor (LACBWR) nuclear power plant. The storage system for DPC is designated MPC-LACBWR. The changes proposed for Amendment No. 6, constitute the third configuration of the NAC-MPC System and include:

  (1) incorporation into the TSC design a single closure lid with a welded closure ring for redundant closure (design features from the MAGNASTOR system [1.3.9 and 1.3.10]);
  (2) modification of the TSC and basket design to accommodate up to 68 LACBWR spent fuel assemblies (36 undamaged Exxon fuel assemblies) and up to 32 damaged fuel cans (in a preferential loading pattern) that may contain undamaged Exxon fuel assemblies, damaged Exxon and Allis Chalmers fuel assemblies and/or fuel debris;
  (3) minor design modifications to the VCC incorporating design features from the MAGNASTOR system that improve operability of the system while adhering to ALARA principles;
  (4) requested the addition of zirconium alloy shroud compaction debris to be stored with undamaged and damaged fuel assemblies;
  (5) to change concrete cask compressive strength from 4,000 to 6,000 psi;
  (6) proposed justification for the 6-foot soil depth as being conservative; and
  (7) other changes to incorporate minor editorial corrections.

  The request, as supplemented, was approved by the NRC in Amendment No. 6 [1.3.1.g] and was effective October 4, 2010.

- **Amendment No. 7:** By application dated November 14, 2017, as supplemented February 12, 2018, NAC requested NRC approval of an amendment to CoC No. 1025 for the NAC-MPC System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. NAC requested approval to identify Technical Specification (TS) A 3.1.6 as not applicable to MPC-LACBWR, removed the Response Surveillance requirement of TS A 5.3 following an off-normal, accident or natural phenomena event, added a finer VCC vent screen mesh for MPC-LACBWR systems, and revised FSAR Sections 3.A.4.4.3.3, 4.A.4, 9.2, and 9.A.3.1. The request, as supplemented, was approved by the NRC in Amendment No. 7 [1.3.1.h] and was effective March 4, 2019.

- **Amendment No. 8**: By application dated February 28, 2018, NAC requested NRC approval of an amendment to CoC No. 1025 for the NAC-MPC System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. NAC requested approval to revise Technical Specification (TS) A 3.1.6 to revise specified required actions and completions, revise TS A 3.2.2 to revise the applicability to ‘Prior to Storage Operations, and deleted TS A 5.3. The request, as supplemented, was approved by the NRC in Amendment No. 8 [1.3.1.i] and was effective March 4, 2019. Due to the close proximity for the approvals Amendments 7 and 8 they were processed together as one rule making package.

1.1.2 NAC-MPC Storage System Loading Overview

General

The NAC-MPC system was specifically designed for older decommissioned nuclear plants having limited facility space and crane capacities. NAC has designed, and NRC has certified three derivatives of the NAC-MPC System: YR-MPC for the Yankee Atomic Electric Company's Yankee Rowe nuclear plant; CY-MPC for the Connecticut Yankee Haddam Neck nuclear plant; and MPC-LACBWR for the Dairyland Power Cooperative La Crosse Boiling Water Reactor nuclear plant. Through September 2012, a total of sixty (60) NAC-MPC systems for SNF storage had been deployed (15 at YR, 40 at CY, and 5 at LACBWR). There are no current plans for additional NAC-MPC System deployments at commercial nuclear plants in the US.

YR-MPC Loading Operations

NAC-MPC System loading operations began at YR with the first system placed into service in May 2002, and the last system placed into service on March 6, 2003. The YR spent fuel assemblies loaded into the YR-MPC were fabricated with both zircaloy and stainless-steel cladding. The lowest heat load system placed into service was fuel loading operation number 7 at 5.71 kW on November 26, 2002, and the highest was number 2 at 8.463 kW on July 17, 2002. The maximum fuel burnup loaded for the YR PWR SFAs was 35,999 MWd/MTU for assembly A739 loaded into TSC loading number 4. Damaged fuel assemblies were pre-loaded into Damaged Fuel Cans (DFCs) prior to loading into the TSC. A total of seven (7) such assemblies were loaded into DFCs and placed into two of the fifteen YR-MPC systems loaded. One (1) RFA was used to accommodate fuel rods from other assemblies.

The YR-MPC units were initially fabricated, constructed, and loaded under NRC CoC No. 1025 revision and amendments as indicated in the second section of Table 1.1-1 below. NAC International subsequently performed an NRC CoC No. 1025 reconciliation in NAC Calculation No. 455-9000, Yankee Atomic Electric Company ISFSI, "NAC-MPC Certificate of Compliance Amendment Reconciliation of Fabrication & Construction of MPC Transportable Storage Canisters and Vertical Concrete Casks, Operational Procedures, and Fuel Contents" [1.3.3]. Revision 0 of the calculation was issued on January 15, 2010 reconciling YR-MPC TSC and VCC Units 1-15 and Damaged Fuel Cans 1-11 to NRC CoC No. 1025, Amendment 5, and Final Safety Analysis Report (FSAR) Revision 7. The YR-MPC Transfer Cask was sold to DPC for MPC-LACBWR loading operations and was not reconciled under the YAEC calculation.

As a result of the reconciliation calculation NAC issued NAC International Supplemental Certificate of Conformance YR-COC-TSC 1-15/VCC 1-15/DFC 1-11, Yankee Atomic Electric Company, January 22, 2010 [1.3.4]. All YR-MPC TSCs, VCCs, and DFCs were certified to be in full compliance with CoC No. 1025, Amendment 5, and NAC-MPC FSAR, Revision 7 as indicated in the first section of Table 1.1-1.

Table 1.1-1 YR-MPC Components CoC Compliance Matrix

YR-MPC System Number	Registered Amendment Usage by the Licensee per 10 CFR 72.212(b)(2)			
	TSC Fabrication	VCC Fabrication	DFC Fabrication	System Loading
1-15	Amendment 5	Amendment 5		Amendment 5
DFC 1-11			Amendment 5	

YR-MPC System Number	NAC-MPC CoC Original As-Fabricated Amendment			
	TSC	VCC	DFC Fabrication	System Loading
TSC 1-5	Amendment 1	Amendment 0		
TSC 6	Amendment 2	Amendment 0		Amendment 2
TSC 7-9	Amendment 1	Amendment 0		Amendment 2
TSC 10-12	Amendment 2	Amendment 0		Amendment 2
TSC 13-14	Amendment 2, with two Exemptions	Amendment 0		Amendment 2
TSC 15	Amendment 1	Amendment 0		Amendment 2
DFC 1-11			Amendment 2	Amendment 2

CY-MPC Loading Operations

NAC-MPC System loading operations began at Connecticut Yankee’s Haddam Neck Nuclear Station with the first system placed into service on May 21, 2004 and the final system placed into service on March 26, 2005. A total of forty (40) CY-MPC units were loaded using two Transfer Casks. The spent fuel assemblies at CY had both zirconium alloy and stainless-steel cladding. The lowest decay heat load was fuel loading operation number’s 31 and 32 at 6.13 kW on February 6 and 9, 2005, and the highest was fuel loading operation number 12 at 12.28 kW on August 18, 2004. The maximum fuel burnup loaded for the CY 15x15 W PWR SFA (W47) was 42,955 MWd/MTU in loading sequence number 18 (TSC No. 12) on October 5, 2004. All damaged fuel assemblies and fuel debris were pre-loaded into Damaged Fuel Cans (DFCs) prior to loading into the TSC. A total of seventy-one (71) damaged fuel assemblies were loaded in DFCs in nineteen (19) of the 40 CY-MPC TSCs loaded.

The Connecticut Yankee NAC-MPC Systems were initially fabricated and constructed under the NRC CoC No. 1025 amendments as indicated in the second section of Table 1.1-2 below. NAC International subsequently performed an NRC CoC No. 1025 reconciliation in NAC Calculation No. 12414-9000, Connecticut Yankee Atomic Power Company ISFSI Spent Fuel Storage Project, “NAC-MPC Certificate of Compliance Amendment Reconciliation of Fabrication & Construction of CY-MPC Transportable Storage Canisters, Vertical Concrete Casks, and Transfer Systems, Operational Procedures, and Fuel Contents” [1.3.5]. Revision 0 was issued on January 15, 2010 reconciling CY-MPC TSC and VCC Units 1-40, Damaged Fuel Cans 1-72 and Transfer Casks 1 and 2 to NRC CoC No. 1025, Amendment 5 and NAC-MPC Final Safety Analysis Report (FSAR), Revision 7.

As a result of the reconciliation calculation NAC issued NAC International Supplemental Certificate of Conformance CY-COC-TSC-VCC-DFC-TFR for Connecticut Yankee Atomic Electric Company, January 22, 2010 [1.3.6]. All CY-MPC TSCs, VCCs, TFRs and DFCs were certified to be in full compliance with CoC No. 1025, Revision 5, and NAC-MPC FSAR, Revision 7 as indicated in the first section of Table 1.1-2.

Table 1.1-2 CY-MPC Components CoC Compliance Matrix

CY-MPC System Number	Registered Amendment Usage by the Licensee per 10 CFR 72.212(b)(2)				
	TSC Fabrication	VCC Fabrication	Transfer Cask	DFC Fabrication	System Loading
CY-MPC 1-40	Amendment 5	Amendment 5			Amendment 5
DFC 1-72				Amendment 5	Amendment 5
CY-MPC Transfer Cask 1 & 2			Amendment 5		Amendment 5

CY-MPC System Number	NAC-MPC CoC Original As-Fabricated Amendment				
	TSC	VCC	Transfer Cask	DFC Fabrication	System Loading
TSC 1-40	Amendment 2	Amendment 2			Amendment 3
DFC 1-11, 13-33, 36-39, and 41-42				Amendment 2	Amendment 3
DFCs 12, 34, 35, 40, and 43-72				Amendment 3	Amendment 3
CY-MPC Transfer Cask 1			Amendment 2		Amendment 3
CY-MPC Transfer Cask 2			Amendment 3		Amendment 3

DPC MPC-LACBWR Loading Operations

NAC-MPC System loading operations began at Dairyland Power Cooperative's (DPC) La Crosse Boiling Water Reactor (LACBWR) Nuclear Station with the first system placed into service in June 2012 and the final system placed into service in September 2012. All of the LACBWR fuel assemblies were manufactured with stainless steel cladding. A total of five (5) MPC-LACBWR systems were loaded using the Yankee Rowe Transfer Cask modified with the addition of two new shield doors and a retaining ring assembly. The lowest decay heat load was fuel loading operation number 5 at 1.586 kW placed into service on September 18, 2012, and the highest decay heat load was fuel loading operation number 3 at 2.773 kW placed into service on August 7, 2012. The maximum fuel burnup loaded for LACBWR SFA (4-47) was 21,532 MWd/MTU in loading sequence number 4 placed into service on August 16, 2012. All damaged fuel assemblies, potential damaged assemblies (Allis Chalmers assemblies), and fuel debris were loaded into Damaged Fuel Cans (DFCs) prior to loading into the TSC. A total one hundred fifty-seven (157) damaged fuel assemblies in DFCs and one (1) fuel debris DFC were loaded in DFCs in all five of the MPC-LACBWR TSCs loaded (up to 32 DFCs per TSC). All MPC-LACBWR systems were loaded and operated in accordance with USNRC CoC No. 1025, Amendment 6 and FSAR Revision 11.

The DPC MPC-LACBWR systems were initially fabricated and constructed under NRC CoC No. 1025 amendments as listed in Table 1.1-3:

Table 1.1-3 MPC-LACBWR Components CoC Compliance Matrix

MPC-LACBWR System Number	Registered Amendment Usage by the Licensee per 10 CFR 72.212(b)(2)				
	TSC Fabrication	VCC Fabrication	Transfer Cask	DFC Fabrication	System Loading
System Number 1-5	Amendment 6	Amendment 6			Amendment 6
DFC 1-165				Amendment 6	Amendment 6
Yankee Rowe Transfer Cask			Amendment 6		Amendment 6

Overall NAC-MPC Operational Experience

No significant storage loading, operational, off-normal or accident events has occurred at any of the three facilities utilizing the NAC-MPC Systems. Lessons learned during initial loading operations at Yankee Rowe, CY, and LACBWR are discussed in Section 3.

1.2 APPLICATION FORMAT AND CONTENT

The NAC-MPC System CoC renewal application format and content of the application are based on the requirements of 10 CFR Part 72.240(c) and the guidance provided in NUREG-1927 [1.3.7]. Table 1.2-1 provides a summary of the section number and headings of the NAC-MPC System CoC renewal application and cross-references to the applicable sections of NUREG-1927 [1.3.7] and 10 CFR Part 72 Regulations.

All changes in the NAC-MPC System that have been previously made without prior NRC approval in accordance with 10 CFR 72.48 have been incorporated in the latest FSAR (Reference 1.3.2.m)

Table 1.2-1 Regulatory Compliance Cross-Reference Matrix

CoC Renewal Application Section Number and Heading	NUREG-1927 Section Number and Heading	10CFR72 Requirement
1. General Information	1. General Information Review	---
1.1 Background Information	---	---
1.1.1 NAC-MPC CoC Amendment History	---	---
1.1.2 NAC-MPC Storage System Loading Overview	---	---
1.2 Application Format and Content	1.4.4 Application Content	§72.240(b), (c)
1.3 References	---	---
2. Scoping Evaluation	2. Scoping Evaluation	---
2.1 Introduction	---	---
2.2 Scoping Methodology	2.4.1 Scoping Process	§72.236
2.3 Scoping Results	---	---
2.4 Description of SSCs and Identification of Intended Function	---	---
2.5 SSCs Within Scope of CoC Renewal	2.4.2 Structures, Systems, and Components Within the Scope of Renewal	§§72.122, 72.236
2.6 SSCs Not Within the Scope of CoC Renewal	2.4.3 Structures, Systems, and Components Not Within the Scope of Renewal	§72.122
2.7 References	---	---
3. Aging Management Review	3. Aging Management Review	---
3.1 Identification of SSC Materials and Environments	3.4.1.1 Identification of Materials and Environments	---
3.1.1 Identification of In-Scope SSC Subcomponent Materials	---	---
3.1.2 Environments	---	---

Table 1.2-1 Regulatory Compliance Cross-Reference Matrix

CoC Renewal Application Section Number and Heading	NUREG-1927 Section Number and Heading	10CFR72 Requirement
3.2 Identification of Aging Effects Requiring Management	3.4.2 Identification of Aging Mechanisms and Effects	§72.236
3.2.1 Possible Aging Effects of MPC TSC and Fuel Basket and Transfer Cask Subcomponents	3.4.1.3 Aging Management Activities	---
3.2.2 Neutron Shielding Materials	---	---
3.2.3 Neutron Poison Materials (Neutron Absorbers)	---	---
3.2.4 Vertical Concrete Cask Subcomponent Materials	---	---
3.2.5 Spent Fuel Assemblies	3.4.1.4 Aging Management Review for Fuel Assemblies	---
3.3 Time-Limited Aging Analyses (TLAA)	3.5 Time-Limited Aging Analysis Evaluation	§72.240(c)(2)
3.3.1 TLAA Identification Criteria		
3.3.2 TLAA Identification Process and Results		
3.3.3 Evaluation and Disposition of Identified TLAAs		
3.4 Aging Management Program	3.6 Aging Management Program	§72.240(c)(3)
3.4.1 Aging Effects Subject to Aging Management	3.6.1.1 Aging Effects Subject to Aging Management	---
3.4.2 Aging Management Program Description	3.6.1.2 Prevention Mitigation, Condition Monitoring, and Performance Monitoring Programs	---
3.5 Tollgate Assessments	3.6.1.10	---
3.6 Fuel Retrievability	---	§72.122(l)
3.7 Operating Experience Review	3.6.1.10	---
3.8 Design Basis Document Review	---	---
3.9 References	---	---
Appendix A – Aging Management Program	Appendix A - Aging Management Programs	---
Appendix B – Time-Limited Aging Analysis	Appendix B – Time Limited Aging Analysis	§72.240(c)
Appendix C – MPC Storage System FSAR Changes	1.4.4 Application Content	§72.240(c)
Appendix D – MPC Storage System Technical Specification Changes	1.4.4 Application Content	§72.240(c)
Appendix E – Pre-Application Test Report	1.4.4 Application Content	§72.240(c)
Appendix F – Design Basis Document Review Report	1.4.4 Application Content	§72.240(c)

1.3 REFERENCES

1.3.1 U.S. Nuclear Regulatory Commission, Certificate of Compliance for Spent Fuel Storage Casks, Model No.: NAC-MPC Certificate No. 1025, Docket No. 72-1025;

  1.3.1.a. NAC-MPC CoC; Initial Issue Amendment 0, Effective April 10, 2000.
  1.3.1.b. NAC-MPC CoC; Amendment No. 1, Effective November 13, 2001.
  1.3.1.c. NAC-MPC CoC; Amendment No. 2, Effective May 29, 2002.
  1.3.1.d. NAC-MPC CoC; Amendment No. 3, Effective October 1, 2003.
  1.3.1.e. NAC-MPC CoC; Amendment No. 4, Effective October 27, 2004.
  1.3.1.f. NAC-MPC CoC; Amendment No. 5, Effective July 24, 2007.
  1.3.1.g. NAC-MPC CoC; Amendment No. 6, Effective October 4, 2010.
  1.3.1.h. NAC-MPC CoC; Amendment No. 7, Effective March 4, 2019.
  1.3.1.i. NAC-MPC CoC; Amendment No. 8, Effective March 4, 2019.

1.3.2 NAC International, Inc., "Final Safety Analysis Report for the NAC-MPC Multi-Purpose Canister System," Docket No. 72-1025;

  1.3.2.a. NAC-MPC System FSAR, Revision 0, May 2000

  1.3.2.b. NAC-MPC System FSAR, Revision 1, February 2002
  1.3.2.c. NAC- MPC System FSAR, Revision 2, November 2002
  1.3.2.d. NAC- MPC System FSAR, Revision 3, March 2004
  1.3.2.e. NAC- MPC System FSAR, Revision 4, November 2004
  1.3.2.f. NAC- MPC System FSAR, Revision 5, October 2005
  1.3.2.g. NAC- MPC System FSAR, Revision 6, November 2006
  1.3.2.h. NAC- MPC System FSAR, Revision 7, November 2008
  1.3.2.i. NAC- MPC System FSAR, Revision 8, February 2009
  1.3.2.j. NAC- MPC System FSAR, Revision 9, November 2010
  1.3.2.k. NAC- MPC System FSAR, Revision 10, January 2014
  1.3.2.l. NAC- MPC System FSAR, Revision 11, April 2018
  1.3.2.m. NAC- MPC System FSAR, Revision 12, April 2019

1.3.3 NAC International, Inc., Calculation No. 455-9000, R0, “NAC-MPC Certificates of Compliance Amendment Reconciliation for the Fabrication & Construction of Yankee MPC Transportable Storage Canisters, Vertical Concrete Casks, Operational Procedures, and Fuel Contents,” dated January 15, 2010.

1.3.4 NAC International, Inc. Supplemental Certificate of Conformance YR-COC-TSC 1-15/VCC 1-15/DFC 1-11, Yankee Atomic Power Company, dated January 22, 2010.

1.3.5 NAC International, Inc., Calculation No. 12414-9000, R0, Connecticut Yankee Atomic Power Company ISFSI Spent Fuel Storage Project, “NAC-MPC Certificate of Compliance Amendment Reconciliation for the Fabrication & Construction of MPC Transportable Storage Canisters, Vertical Concrete Casks and Transfer Casks, Operational Procedures, and Fuel Contents,” dated January 15, 2010.

1.3.6 NAC International, Inc. Supplemental Certificate of Conformance CY-COC-TSC-VCC-DFC-TFR for Connecticut Yankee Atomic Power Company, dated January 22, 2010.

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

1.3.8 NEI 14-03, “Guidance for Operations Based Aging Management for Dry Cask Storage,” Revision 2, December 2016.

1.3.9 MAGNASTOR Final Safety Analysis Report, Revision 9, August 2017.

1.3.10 U.S. Nuclear Regulatory Commission, Certificate of Compliance for Spent Fuel Storage Casks, Model No.: MAGNASTOR, Certificate No. 1031, Docket No. 72-1031; Amendment No. 7, Effective August 21, 2017.

2.0 SCOPING EVALUATION

2.1 INTRODUCTION

The NAC-MPC System CoC renewal methodology follows NUREG-1927 [2.7.7] and NEI 14-03 [2.7.4]. The 10 CFR Part 72 CoC renewal process adopts the regulatory philosophy of 10 CFR Part 54. This philosophy is summarized in the two principles of CoC renewal from 10 CFR Part 54 Final Rule Statements of Consideration [2.7.8] which are re-stated below:

> *"The first principle of CoC renewal was that, with the exception of age-related degradation unique to CoC renewal and possibly a few other issues related to safety only during the period of extended operations of nuclear power plants, the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provides and maintains an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security. Moreover, consideration of the range of issues relevant only to extended operation led the Commission to conclude that the detrimental effects of aging is probably the only issue generally applicable to all plants. As a result, continuing this regulatory process in the future will ensure that this principle remains valid during any period of extended operation if the regulatory process is modified to address age-related degradation that is of unique relevance to CoC renewal. …"*
>
> *"The second and equally important principle of CoC renewal holds that the plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent during the original licensing term. This principle would be accomplished, in part, through a program of age-related degradation management for systems, structures, and components that are important to CoC renewal…"*

Based on these principles, CoC renewal is not intended to impose requirements beyond those that were met by the storage system and facility when it was initially certified by the NRC. Therefore, the current licensing basis for the NAC-MPC System will be carried forward through the renewed 40-year CoC renewal period.

The scoping process involves identification of the SSCs of the NAC-MPC System that are within the scope of CoC renewal, and thus require evaluation for the effects of aging. A description of the scoping process is provided in Section 2.2.

2.2 SCOPING METHODOLOGY

The first step in the CoC renewal process involves the identification of the in-scope NAC-MPC System SSCs. This is done by evaluating the SSCs that comprise the NAC-MPC System against the following scoping criteria provided in NUREG-1927 [Reference 2.7.7].

1. *They are classified as important to safety, as they are relied on to do one of the following:*
   - *Maintain the conditions required by the regulations, license, or CoC to store spent fuel safely*
   - *Prevent damage to the spent fuel during handling and storage*
   - *Provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public*

   These SSCs ensure that important to safety functions (ITS) are met for:
   - Subcriticality (CR),
   - Thermal/heat removal (TH),
   - Confinement (CO),
   - Radiation shielding (SH),
   - Structural integrity (SR), and
   - Retrievability (RE)

2. *They are classified as <u>not</u> important to safety (NITS) but, according to the licensing basis, their failure could prevent fulfillment of a function that is important to safety, or their failure as support SSCs could prevent fulfillment of a function that is important to safety.*

Any NAC-MPC System SSC that meets either scoping criterion 1 or 2 above is considered within the scope of license renewal (in-scope), and the function(s) it is required to perform during the extended term is identified. The results of the scoping evaluation are presented in Section 2.3

In accordance with NUREG-1927 [2.7.7] the NAC-MPC System CoC renewal is based on the continuation of the Current Licensing Basis (CLB) throughout the period of extended operation (PEO) and maintenance of the intended safety functions of SSC ITS. Thus, the current licensing basis is reviewed to determine those SSCs with intended functions that meet either scoping criterion 1 or 2, as defined above. The following documents comprise the current licensing basis for the NAC-MPC System.

- NAC-MPC System FSAR [Reference 2.7.1.a thru 2.7.1.m]
- CoC No. 1025 [Reference 2.7.2.a thru 2.7.2.i]

The FSAR provides a description of the cask system, SSCs and their functions, including safety classifications as established by the safety analysis. The applicable NAC-MPC System License Drawings utilized in the scoping process and contained in the approved FSARs are listed in Tables 2.2-1, 2.2-2, and 2.2-3 for the YR-MPC, CY-MPC, and MPC-LACBWR, respectively. The CoC and associated Technical Specifications, govern the storage of irradiated nuclear fuel in the NAC-MPC System, and the transfer of irradiated fuel to and from the spent fuel pool (SFP) and the cask storage pad. Additionally, the Safety Evaluation Report [Reference 2.7.3.a thru 2.7.3.i], which summarizes the results of the NRC staff's safety review of the original licensing, and the Safety Evaluation Reports (SERs) associated with subsequent amendments were considered in the CoC renewal scoping process.

2.3 SCOPING RESULTS

The SSCs comprising the NAC-MPC System are identified in Table 2.3-1, Scoping Results. Those SSCs meeting scoping Criterion 1 or 2 are identified in the table as being within the scope of the CoC renewal.

As indicated in Table 2.3-1, the Transportable Storage Canister (TSC), Vertical Concrete Cask (VCC), Transfer Cask (TFR), and Spent Fuel Assemblies (SFA) were determined to be ITS and therefore, within the scope of CoC renewal and requiring further review in the aging management review process. Although not within the scope of the CoC renewal, the ISFSI Pad has been identified to be ITS by some the General Licensees and requiring further review for aging management. The aging management of ISFSI Pads identified as ITS will be managed by the General Licensee on a site-specific basis.

SSCs determined to be NITS and not meeting Criterion 2 include Fuel Transfer Equipment, Ancillary Operating Systems, Temperature Monitoring Equipment, ISFSI Security Equipment, and other utility services or equipment. At some ISFSIs the storage pad is considered a site-specific ITS structure and will be evaluated on a site-specific basis.

Subcomponents that are identified as having an intended passive function that supports the passive safety function of its associated SSC are part of the aging management review under Criterion 1. The intended functions of the subcomponents are categorized as one or more of the following safety functions:

1. Subcriticality (CR)
2. Thermal/Heat Removal (TH)
3. Confinement (CO)
4. Radiation Shielding (SH)
5. Structural Integrity (SR)
6. Retrievability (RE)

In addition, SSC subcomponents that do not directly support a passive safely function of the SSC are reviewed to identify whether these subcomponents' failure impact another SSC subcomponents' passive safety function and are identified as requiring aging management review under Criterion 2. The results of these reviews are discussed in Section 2.5 below and associated SSC subcomponent tables.

2.4 DESCRIPTION OF SSCs AND IDENTIFICATION OF INTENDED FUNCTION

2.4.1 Description of SSC

The NAC-MPC System is provided in three configurations, the YR-MPC, the CY-MPC, and the MPC-LACBWR, which have similar components and operating features, but different physical dimensions, weights, fuel contents, and storage capacities. All configurations are designed to provide long-term storage and subsequent transport of the stored spent fuel in the TSC using the certified NAC-STC transport cask system. During long-term storage, the NAC-MPC System is designed to provide an inert environment; passive shielding, cooling, and criticality control; and, a confinement boundary closed by welding. The structural integrity of the system precludes the release of contents in any of the design basis normal conditions and off-normal or accident events, thereby assuring public health and safety during use of the system.

The TSC provides the confinement pressure boundary, heat transfer, criticality control and structural integrity for the safe storage of the contained SFAs. The TSC is stored in the central cavity of the VCC. The VCC provides radiation shielding and structural protection for the TSC and contains internal air flow paths that allow the decay heat from the TSC contents to be removed by natural air circulation around the TSC shell. The principal components identified as potential in-scope SSCs of the NAC-MPC System are:

- TSC (YR-MPC; CY-MPC; and MPC-LACBWR) with PWR or BWR Fuel Basket (and Damaged Fuel Cans [DFCs])
- VCC (YR-MPC; CY-MPC; and MPC-LACBWR)
- Transfer Cask (TFR) (YR-MPC as modified and transferred/sold to MPC-LACBWR, and; CY-MPC) and Transfer Adapter
- Spent Fuel Assemblies (SFAs)
- Fuel Transfer and Auxiliary Equipment (e.g., lift yoke, vertical cask transporter, air pads, heavy haul transfer trailer, vacuum drying and helium back-fill system with a helium mass spectrometer leak detector, welding equipment)
- VCC Temperature Monitoring System
- ISFSI Storage Pad
- ISFSI Security Equipment

License Drawings of the NAC-MPC System components and equipment are provided in the FSAR that correspond with the initial CoC and all approved CoC amendments. Tables summarizing the components on the FSAR License Drawings associated with the initial CoC and all subsequent amendments is provided in Tables 2.2-1, 2.2-2, and 2.2-3 for YR-MPC, CY-MPC and MPC LACBWR, respectively. Descriptions of the SSCs are provided in Section 2.4.2 through 2.4.8

2.4.2 Transportable Storage Canister (TSC) and Fuel Basket

The NAC-MPC System TSC and integral fuel baskets are described in Sections 1.2.1.1 (YR-MPC and CY-MPC) and 1.A.2.1.1 (MPC-LACBWR) of the NAC-MPC FSAR [2.7.1.a thru 2.7.1.m]. Three unique TSC designs are included in the NAC-MPC System to accommodate the three types of SFAs (YR and CY PWR, and LACBWR fuel assemblies). The three TSC designs differ in length, closure design and shell and bottom plate thicknesses. All three TSCs have identical nominal outside diameters. The NAC-MPC TSC is designed to be transported in the NAC-STC Transport Cask and transport conditions establish the design basis load conditions for the TSC, except for canister lifting. The transport load conditions produce higher stresses in the canisters than would be produced by the storage load conditions. Consequently, the canister designs are conservative with respect to storage conditions. The evaluation of the canister for transport conditions is documented in the Safety Analysis Report for the NAC Storage Transport Cask (NAC-STC), Docket No. 71-9235 [2.7.5], and approved in NRC CoC No. 71-9235 [2.7.6].

The YR-MPC and CY-MPC TSC assemblies consist of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The cylindrical shell, the bottom plate and lids constitute the confinement boundaries. The baskets feature the NAC-patented poison tubes and stacked disk design with heat transfer disks. The baskets are analyzed using the ANSYS computer code to demonstrate that it can withstand the horizontal drop loads without deforming in a way that damages or constrains a fuel assembly to prevent retrieval.

The fuel basket designs are right-circular cylinder configurations with either 24, 26, or 36 fuel tubes laterally supported by a series of support disks, which are retained by spacers on radially located tie rods. Connecticut Yankee fuel is stored in either a 24- or 26-assembly basket configuration, while Yankee Class fuel is stored in the 36-assembly configuration. Eight tie rods are used in the YR-MPC basket design. Six tie rods are used in the CY-MPC basket. The support disks are stainless steel (17-4 PH) with holes for the poison fuel tubes or damaged fuel cans. YR-MPC fuel baskets have 22 support disks and CY-MPC fuel baskets have 28 support disks. The basket top and bottom weldments are fabricated from Type 304 stainless steel. The tie rods and spacer sleeves are also fabricated from Type 304 stainless steel. The fuel assemblies are contained in fuel tubes or DFCs.

There are three YR-MPC basket configurations that incorporate two fuel tube configurations and a damaged fuel can configuration. The tubes are fabricated from 18-gauge Type 304 stainless steel sheet. The standard YR-MPC fuel tube has a square interior cross-section of 7.8 inches and is encased with BORAL sheets on all four outside surfaces of the fuel tube. The enlarged YR-MPC fuel tube has a square interior cross-section of 8.0 inches but does not have exterior BORAL sheets on the sides. These larger cross-section fuel tubes can accommodate fuel assemblies that exhibit slight physical effects (e.g., twist, bow) that could preclude loading in the smaller cross-section standard fuel tubes. The enlarged fuel tubes are restricted to the four corner positions of the basket. When installed, the standard and enlarged fuel tubes are captured between the top and bottom weldments of the fuel basket.

The three YR-MPC basket configurations accommodate 36 standard fuel tubes, 32 standard fuel tubes and four enlarged fuel tubes at the four basket corner positions or 32 standard fuel tubes and four damaged fuel cans at the four basket corner positions. The basket configurations are not interchangeable.

There are three CY-MPC basket configurations that incorporate two fuel tube configurations and a damaged fuel can configuration. The standard CY-MPC fuel tube has a square interior cross-section of 8.72 inches and is encased with BORAL sheets on all four outside surfaces of the fuel tube. The enlarged CY-MPC fuel tube has a square interior cross-section of 9.12 inches and is encased with BORAL sheets on all four outside surfaces of the fuel tube. These larger cross-section fuel tubes can accommodate fuel assemblies that exhibit slight physical effects (e.g., twist, bow) that could preclude loading in the smaller cross-section standard fuel tubes. The enlarged fuel tubes are restricted to the four corner positions of the basket. When installed, the standard and enlarged fuel tubes are captured between the top and bottom weldments of the fuel basket.

The three CY-MPC basket configurations accommodate 24 or 26 standard fuel tubes or 20 or 22 standard fuel tubes and four enlarged fuel tubes at the four basket corner positions that can also accommodate four damaged fuel cans at the four basket corner positions. The basket configurations are not interchangeable.

The damaged fuel can designs for both YR-MPC and CY-MPC do not have exterior BORAL sheets on the sides and are restricted to the four corner positions of the basket. The damaged fuel can is closed on its bottom end by a stainless steel bottom plate having screened openings. After loading, the can is closed on its top end by a stainless steel lid that also has screened openings. The top plate and can body incorporate lifting fixtures that allow movement of the loaded DFC, if necessary, and installation and removal of the DFC lid. The DFC extends through the bottom and top weldments of the basket, and is captured between the shield lid configured for damaged fuel cans and the canister bottom plate. The screened openings in the damaged fuel can lid and bottom plate allow the filling, draining and vacuum drying of the DFC and stored SFA, but preclude the release of gross particulate matter to the canister interior.

To permit full access to the enlarged fuel tubes, the corner positions of the top and bottom weldments used in the damaged fuel can basket configurations for both YR-MPC and CY-MPC are also enlarged. However, the enlarged fuel tubes remain captured between the basket top and bottom weldments.

To permit removal, if necessary, of the DFC, the top and bottom weldment openings in the four corner positions of the DFC basket configurations for both the YR-MPC and CY-MPC are sized to allow the DFC to be inserted or removed with the basket assembled. Consequently, the DFC is not captured between the weldments and is retrievable.

Since the standard fuel tube with attached BORAL, the enlarged fuel tube with or without BORAL, and the DFC without BORAL have the same external dimensions, the support disks and heat transfer disks used in the YR-MPC and CY-MPC basket configurations are identical for each design.

The heat transfer disks are aluminum plates with holes for the fuel tubes or DFCs. The heat transfer disks are spaced midway between the support disks and are the primary path for conducting the heat from the fuel assemblies to the canister wall. Holes in the heat transfer disks for the tubes, damaged fuel cans, and tie rods are sized to accommodate thermal expansion occurring after the fuel is placed into the basket. YR-MPC fuel baskets have 14 heat transfer disks and CY-MPC fuel baskets have 27 heat transfer disks.

The fuel basket tube-and-disk design provides the structural integrity to maintain the spent fuel in a subcritical configuration during normal operations and the hypothetical accident events, even if optimum moderator condition and fresh fuel are assumed. With the most reactive fuel, the fuel basket maintains $k_{eff} \leq 0.95$. Subcriticality is assured assuming fresh fuel loading and no soluble boron in the spent fuel pool water during fuel loading operations.

The YR-MPC and CY-MPC TSCs are designed to facilitate filling with water and subsequent draining and drying. Each fuel tube is supported by the basket bottom weldment, ensuring free flow of water between the inner tube regions and the bottom of the canister. The top lid and bottom plate of the damaged fuel can incorporate screened openings to allow water to fill and drain during loading and canister closure operations. Each of the support and heat transfer disks also has three holes to supplement the flow of water between disks. In addition, the bottom weldment is positioned by supports above the bottom of the canister to facilitate water flow to the drain line.

The canister shell is fabricated from ⅝-inch thick Type 304L stainless steel rolled plate, joined at its edges by a full penetration weld, which is radiographed. The bottom closure is a Type 304L stainless steel plate joined to the canister shell by a full penetration weld, which is ultrasonically examined. The bottom plate of the YR-MPC canister is 1-inch thick. The bottom plate of the CY-MPC canister is 1.75-inch thick. The stainless-steel material was selected to minimize the potential for any adverse chemical reactions in the spent fuel pool. The design of the 5-inch thick shield lid and 3-inch thick structural lid allows a redundant confinement boundary at the top of the canister. A backing ring, also called a spacer ring, is installed on the structural lid to support the structural lid-to-canister shell weld. Each lid weld is inspected using liquid penetrant examination on the root and final or root, intermediate, and final passes.

The shield lid for the YR-MPC TSC used with the damaged fuel can basket configuration incorporates four machined recesses in the underside of the lid to accommodate the damaged fuel cans. The shield lid configured for damaged fuel cans cannot be used interchangeably with other YR-MPC TSC basket configurations.

The vent and drain ports through the shield lid allow the inner cavity to be drained, evacuated, and backfilled with helium to provide an inert atmosphere for long-term dry storage of the SFAs. The drain port is equipped with a quick disconnect fitting and a drain tube that extends nearly to the bottom of the canister. The vent port extends to the underside of the shield lid and is equipped with a quick disconnect fitting used for vacuum drying and helium backfilling. After draining, drying, backfilling, and testing operations are complete, port covers are installed and welded to the shield lid to seal the penetration.

The third NAC-MPC TSC configuration is the MPC-LACBWR TSC which is designed to accommodate up to 68 LACBWR spent fuel assemblies, including up to 32 damaged fuel cans. The MPC-LACBWR TSC assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a closure lid, closure ring and two redundant sets of penetration port covers. The cylindrical shell, plus the bottom plate, closure lid and inner port covers constitute the confinement boundary. The fuel basket design and configuration are similar to and based on the directly loaded fuel basket design used in the certified NAC-STC, NAC-UMS and NAC-MPC storage and transport systems. The MPC-LACBWR basket features the NAC-patented poison tubes and stacked disk design with heat transfer disks. The basket was analyzed using the ANSYS computer code to demonstrate that it can withstand the horizontal drop loads without deforming in a way that damages or constrains a fuel assembly.

The MPC-LACBWR fuel basket design is a right-circular cylinder configuration with 68 fuel tubes laterally supported by a series of support disks, which are retained by spacers on radially located tie rods. Damaged fuel cans may be placed in 32 peripheral oversized fuel tubes. Eight tie rods are used in the MPC-LACBWR basket design. The support disks are stainless steel (17-4 PH) with standard and oversized holes for the poison fuel tubes and damaged fuel cans. The first top and bottom support disks are thicker (1-¾ and ¾ inch respectively) than the 24 intermediate support disks (⅝ inch) to accommodate postulated rubblized fuel in the 32 damaged fuel cans. The basket top and bottom weldments are fabricated from Type 304 stainless steel. The tie rods and spacer sleeves are also fabricated from Type 304 stainless steel. The fuel assemblies are contained in fuel tubes. The MPC-LACBWR fuel tubes are fabricated from Type 304 stainless steel with stainless steel-clad covered BORAL sheets on defined outside surfaces of the fuel tube. The BORAL provides criticality control in the basket.

The MPC-LACBWR fuel tubes are fabricated from 18-gauge Type 304 stainless steel sheet. The standard fuel tube has a square interior cross-section of 5.75 inches and supports a clad covered BORAL sheet on defined outside surfaces of the fuel tube. The enlarged fuel tube has a square interior cross-section of 6.0 inches and supports a clad covered BORAL sheet on three or four sides. Enlarged fuel tubes with BORAL sheets on three sides have an aluminum sheet on the fourth side to provide a symmetric interface between the fuel tube and the top basket support disk. These larger cross-section fuel tubes can accommodate damaged fuel cans and fuel assemblies that exhibit slight physical effects (e.g., twist, bow) that could preclude loading in the smaller cross-section standard fuel tubes. The enlarged fuel tubes are located in the 32 periphery fuel cell positions of the basket. When installed, the standard and enlarged fuel tubes are captured between the top and bottom weldments of the fuel basket.

The MPC-LACBWR damaged fuel can is similar to a fuel tube without exterior BORAL sheets on the sides and is closed on its bottom end by a stainless steel bottom plate having screened openings. After loading, the can is closed on its top end by a stainless steel lid that also has screened openings. The top plate and can body incorporate lifting fixtures that allow movement of the loaded DFC, and installation and removal of the can lid. The DFC extends through the bottom and top weldments of the basket, and is captured between the closure lid and the canister bottom plate. The DFC lid is held in place by the closure lid. The screened openings in the DFC lid and bottom plate allow the filling, draining and vacuum drying of the damaged fuel can, but preclude the release of gross particulate matter to the canister interior.

The 14 heat transfer disks are aluminum plates with holes for the standard and enlarged fuel tubes. The heat transfer disks are spaced midway between the support disks and are the primary path for conducting the heat from the fuel assemblies to the canister wall. Holes in the heat transfer disks for the tubes, damaged fuel cans, and tie rods are sized to accommodate thermal expansion occurring after the fuel is placed into the basket.

The fuel basket tube-and-disk design provides the structural integrity to maintain the spent fuel in a subcritical configuration during normal operations and the hypothetical accident events, even if optimum moderator condition and fresh fuel are assumed. With the most reactive fuel, the fuel basket maintains $k_{eff} \leq 0.95$. Subcriticality is assured assuming fresh fuel loading and no soluble boron in the spent fuel pool water during fuel loading operations.

The MPC-LACBWR TSC assembly is designed to facilitate filling with water and subsequent draining and drying. Each fuel tube is supported by the basket bottom weldment, ensuring free flow of water between the inner tube regions and the bottom of the canister. The top lid and bottom plate of the damaged fuel can incorporate screened openings to allow water to fill and drain during loading and canister closure operations. In addition, the bottom weldment is positioned by supports above the bottom of the canister to facilitate water flow to the drain line.

The MPC-LACBWR TSC is fabricated from ½-inch-thick dual certified Type 304/304L stainless steel rolled plate, joined at its edges by a full penetration weld, which is radiographed. The bottom plate is a 1.25-inch-thick Type 304/304L stainless steel plate joined to the canister shell by a full penetration weld, which is ultrasonically examined. The design of the 7-inch thick closure lid and closure ring with dual redundant port covers provides a redundant confinement boundary at the top of the canister. The closure lid weld to the canister shell is inspected using liquid penetrant examination on the root, intermediate, and final passes.

The MPC-LACBWR closure lid design includes a 4-inch-thick, 38.3-inch-square aluminum spacer plate attached to the underside of the lid to limit axial movement of the fuel assemblies placed in the 36 basket locations that do not contain damaged fuel cans. Axial movement of the damaged fuel cans is limited by the position of the closure lid bottom surface.

The vent and drain ports through the closure lid allow the inner cavity to be drained, evacuated, and backfilled with helium to provide an inert atmosphere for long-term dry storage. The drain port is equipped with a quick disconnect fitting and a drain tube that extends nearly to the bottom of the canister. The vent port extends to the underside of the closure lid and is equipped with a quick disconnect fitting used for vacuum drying and helium backfilling. After draining, drying, backfilling, and testing operations are complete, port covers are installed and welded to the closure lid to seal the penetration. Leak testing is performed on both inner port cover welds followed by installation of a second redundant port cover for each port.

2.4.3 Vertical Concrete Cask (VCC)

The NAC-MPC System VCC is the storage overpack for the TSC and is provided in three configurations. The VCC designs are described in Sections 1.2.1.2 and 1.A.2.1.2 of the NAC-MPC System FSAR [2.7.1.a thru 2.7.1.m]. The YR-MPC and CY-MPC VCC designs are similar, and the MPC-LACBWR VCC design incorporates features from the certified MAGNASTOR System.

The YR-MPC and CY-MPC VCCs are the storage overpacks for the YR-MPC and CY-MPC TSCs respectively. The NAC-MPC VCCs provide structural support, shielding, protection from environmental conditions, and natural convection cooling of the canister during long-term storage, and are essentially identical in function but with different overall dimensions to accommodate the YR-MPC and CY-MPC TSCs. The NAC-MPC VCC is a reinforced concrete (Type II Portland cement) structure with a structural steel inner liner. The concrete wall and steel liner provide neutron and gamma radiation shielding. Inner and outer reinforcing steel (rebar) assemblies are contained within the concrete. The reinforced concrete wall provides the structural strength to protect the canister and its contents in natural phenomena events such as tornado wind loading and wind-driven missiles. The storage cask incorporates reinforced chamfered corners at the edges to facilitate construction.

The YR-MPC VCC base plate weldment is covered with a ¼-inch-thick stainless-steel plate backed by a silicone foam insulating material to prevent contact between the stainless-steel canister and the carbon steel pedestal, and to limit heat dissipation from the TSC baseplate to the pedestal. The CY-MPC VCC base weldment base plate is covered with a ¼-inch-thick stainless-steel plate to prevent contact between the stainless-steel canister and the carbon steel pedestal. The storage cask has an annular air passage to allow the natural circulation of air around the canister to remove the decay heat from the spent fuel. The air inlet and outlet vents are steel-lined penetrations that take nonplanar paths to the concrete cask cavity to minimize radiation streaming. The decay heat is transferred from the fuel assemblies to the fuel tubes or damaged fuel can in the fuel basket and through the heat transfer disks to the canister wall. Heat flows by radiation and convection from the canister wall to the air circulating through the concrete cask annular air passage and is exhausted through the air outlet vents. This passive cooling system is designed to maintain the peak cladding temperature of both stainless steel and zirconium alloy clad fuel well below acceptable limits during long-term storage. This design also maintains the bulk concrete temperature below 150°F and localized concrete temperatures below 200°F in normal operating conditions. The YR-MPC VCC inlets are provided with removable VCC inlets supplemental shields, which reduce the local dose adjacent to the inlets for ALARA purposes without reducing the thermal performance of the YR-MPC VCC.

The top of the Yankee-MPC and CY-MPC VCCs are closed by a shield plug and lid. The shield plug for the Yankee-MPC VCC is approximately 5 inches thick and incorporates carbon steel plate as gamma radiation shielding and NS-4-FR as neutron radiation shielding. A carbon steel lid that provides additional gamma radiation shielding is installed above the shield plug. For the CY-MPC VCC, the shield plug is similar to the Yankee-MPC VCC except the neutron shielding may be either NS-4-FR or NS-3. The VCC shield plug and lid reduce skyshine radiation and provide a cover and seal to protect the canister from the environment and postulated tornado missiles.

The MPC-LACBWR VCC is the storage overpack for the MPC-LACBWR TSC. It provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the TSC during long-term storage. The MPC-LACBWR VCC is a reinforced concrete (Type II Portland cement) structure with a structural steel inner liner. The concrete wall and steel liner provide neutron and gamma radiation shielding. Inner and outer reinforcing steel (rebar) assemblies are contained within the concrete. The reinforced concrete wall provides the structural strength to protect the canister and its contents in natural phenomena events such as tornado wind loading and wind-driven missiles. The MPC-LACBWR VCC incorporates reinforced chamfered corners at the edges to facilitate construction. The MPC-LACBWR VCC base weldment base plate is covered with a ¼-inch-thick stainless-steel plate to prevent contact between the stainless-steel canister and the carbon steel pedestal.

The MPC-LACBWR VCC has an annular air passage to allow the natural circulation of air around the canister to remove the decay heat from the spent fuel. The air inlets and outlets are steel-lined penetrations that take nonplanar paths from the concrete cask cavity to minimize radiation streaming. The decay heat is transferred from the fuel assembly to the fuel tube or damaged fuel can and fuel tube in the fuel basket and through the heat transfer disks to the canister wall. Heat flows by radiation and convection from the canister wall to the air circulating through the concrete cask annular air passage and is exhausted through the air outlets. This passive cooling system is designed to maintain the peak cladding temperature well below acceptable limits during long-term storage. This design also maintains the bulk concrete temperature below 150°F and localized concrete temperatures below 200°F in normal operating conditions. The MPC-LACBWR VCC inlets are fitted with welded pipes to provide additional local shielding in areas adjacent to the inlets for ALARA purposes without reducing the thermal performance of the MPC-LACBWR VCC (similar to the YR-MPC VCC).

The top of the MPC-LACBWR VCC is closed by a lid with integral radiation shield. The radiation shield is approximately 8-inch thick concrete encased in a carbon steel shell extending into the cask cavity from the bottom surface of the 1.5-inch-thick carbon steel lid. This is different than the design for YR-MPC and CY-MPC VCCs.

Fabrication of the NAC-MPC VCCs involve no unique or unusual forming, concrete placement, or reinforcement requirements. The concrete portion of the MPC-LACBWR VCC is constructed by placing concrete between a reusable, exterior form and the inner metal liner. Reinforcing bars are placed near the inner and outer concrete surfaces to provide structural integrity. The inner liner and base of the MPC-LACBWR VCC are shop fabricated. Radiation shielding is installed in the MPC-LACBWR VCC air inlets to reduce dose rates local to the air inlets at the base of the cask.

2.4.4 Transfer Cask (TFR) and Transfer Adapter

The NAC-MPC System Transfer Cask (TFR) is primarily a lifting device described in Section 1.2.1.3 of the NAC-MPC FSAR [2.7.1.a thru 2.7.1.m]. The TFR is used to lift and move the TSC assembly and provides biological shielding when it contains a loaded canister. The TFR is used for the vertical transfer of the TSC between workstations and the VCC, or transport cask. A Transfer Adapter is utilized with the TFR to facilitate positioning and orientation on the VCC or Transport Cask, to provide additional shielding during TSC transfer, and to remotely operate the TFR shield doors.

The basic design of the two NAC-MPC TFRs are similar, with the CY-MPC TFR being approximately 30 inches longer and 2.5 inches larger in external diameter than the YR-MPC TFR. Following utilization at Yankee Rowe, the YR-MPC TFR was sold to and refurbished by DPC for use in loading and transferring the MPC-LACBWR systems. The refurbishment included fabrication of two new shield doors, a retaining ring assembly, and re-load testing of the TFR to ANSI N14.6 requirements.

The NAC-MPC TFRs are multiwall (steel/lead/NS-4-FR neutron shield/steel) designs, which limits the average contact radiation dose rate. The TFR designs incorporate a top retaining ring, which is bolted in place preventing a loaded canister from being inadvertently removed through the top of the transfer cask. The TFR has two retractable bottom shield doors. During TSC/TFR loading operations, the doors are closed and secured by lock bolts/lock pins, so they cannot inadvertently open. During TFR unloading operations, the doors are retracted using hydraulic cylinders installed on the Transfer Adapter to allow the canister to be lowered into a concrete cask for storage or into a transport cask. The Transfer Adapter also provides additional shielding for operational staff during TSC transfer operations.

To qualify the transfer casks as a heavy lifting device, they are designed, fabricated, and proof-load tested to the requirements of NUREG-0612 [2.7.9] and ANSI N14.6 [2.7.10]. Maintenance is performed in accordance with site-specific procedures that meet the requirements of NUREG-0612 and the NAC-MPC System Operating Manuals.

To minimize potential contamination of the TSC and TFR interior surfaces during loading operations in the spent fuel pool, clean water is circulated in the gap between the TFR interior surface and the TSC exterior surface using fill and drain lines located in the top and base of the transfer cask walls. The clean water flow precludes the intrusion of pool water when the TFR/TSC is submerged. Clean water is processed or filtered pool water, or any water external to the spent fuel pool that is compatible.

Exposed surfaces of the TFRs, other than the load-bearing surfaces of the trunnions and the bottom door rails, are coated with approved coating systems to protect the carbon steel and to provide a smooth surface to facilitate decontamination.

2.4.5 Spent Fuel Assemblies (SFAs)

The spent fuel assemblies loaded in the NAC-MPC Systems have specific safety functions which result in the assemblies being defined as ITS SSCs. These safety functions include maintaining the fissile material geometry, maintaining confinement of the radioactive materials within the fuel cladding, and maintaining the ability to retrieve the fuel assemblies.

The NAC-MPC System is provided in three configurations. The YR-MPC for Yankee Class spent fuel, the CY-MPC for Connecticut Yankee spent fuel, and MPC-LACBWR for Dairyland Power Cooperative La Crosse Boiling Water Reactor (LACBWR) spent fuel. The design criteria for the spent fuel stored in the YR-MPC and CY-MPC configurations are described in Section 2.1 of the NAC-MPC System FSAR [2.7.1.a thru 2.7.1.m]. The design criteria for the spent fuel stored in the MPC-LACBWR configuration are described in Section 2.A.1 of the NAC-MPC System FSAR [2.7.1.a thru 2.7.1.m].

The YR-MPC is designed to store up to 36 Yankee Class spent fuel assemblies including up to 4 damaged fuel cans. The Connecticut Yankee CY-MPC is designed to store up to 26 Connecticut Yankee spent fuel assemblies and is provided with either a 26-assembly or a 24-assembly basket. Both CY-MPC baskets can include up to 4 damaged fuel cans. The Dairyland Power Cooperative La Crosse Boiling Water Reactor (LACBWR) is designed to store up to 68 LACBWR spent fuel assemblies, including up to 32 LACBWR damaged fuel cans. The spent fuel assemblies stored in all configurations are delineated by various factors including manufacturer, type, enrichment, burnup, cool time, and cladding material.

The Yankee Class fuel consists of two types of 16x16 arrays, designated A and B. The Type A assembly incorporates a protruding corner of fuel rods while the Type B assembly omits one corner of the fuel rods. Connecticut Yankee spent fuel assemblies are 14x14 PWR Westinghouse-type fuel assemblies. The Connecticut Yankee spent fuel assemblies and the Yankee class fuel assemblies include both stainless steel and zirconium alloy fuel rod cladding.

The LACBWR fuel contents consists of two types, Allis Chalmers and Exxon fuel assemblies. LACBWR fuel assemblies are comprised of 10x10 array of rods, with Allis Chalmers fuel containing 100 fuel rods and Exxon fuel containing 96 fuel rods and four inert rods. All LACBWR fuel assemblies are stainless steel clad. LACBWR fuel assembly shrouds (channels) were removed from the spent fuel assemblies prior to dry fuel storage.

All damaged fuel and fuel debris for all authorized NAC-MPC SNF is required to be placed in a damaged fuel can (DFC) during storage in the TSC. There are no high burnup (HBU) fuel assemblies currently loaded or planned to be loaded in a NAC-MPC System.

2.4.6 Fuel Transfer and Auxiliary Equipment

The fuel transfer and auxiliary equipment necessary for NAC-MPC System loading and ISFSI operations (e.g., lifting yoke, air-pallets, heavy haul trailer, vacuum drying and helium backfill system, welding equipment, weld inspection equipment, drain pump equipment, and helium leak detection equipment) are not included as part of the NAC-MPC System certified in NRC Certificate of Compliance for the NAC-MPC System and as such, are not described in detail in the NAC-MPC System FSAR [2.7.1.a thru 2.7.1.m]. General descriptions of the fuel transfer and auxiliary equipment are provided in Section 1.2.1.5, and in Table 8.1.1-1 of Chapter 8 Operating Procedures in the NAC-MPC System FSAR. Some of the fuel transfer and auxiliary equipment is also depicted in the operational schematics shown in Chapter 1 figures of the NAC-MPC System FSAR.

2.4.7 VCC Temperature Monitoring System

The NAC-MPC System's temperature monitoring system is one method authorized to verify the continued operability of the VCC heat removal system, although it is not part of the system authorized by the NRC in the NAC-MPC System CoC [2.7.2.a thru 2.7.2.i], and as such, is not described in detail in the NAC-MPC FSARs [2.7.1.a thru 2.7.1.m]. The VCC heat removal system is designed to maintain stored fuel cladding and NAC-MPC System components within allowable temperature limits for a period exceeding 24 hours to allow corrective actions to be taken to re-establish operability of the VCC heat removal system.

2.4.8 ISFSI Storage Pad

The NAC-MPC System ISFSI storage pad is not part of the NAC-MPC System approved by the NAC-MPC System CoC [2.7.2.a thru 2.7.2.i], and as such, is not described in detail in the NAC-MPC System FSAR [2.7.1.a thru 2.7.1.m] . The concepts of the YR-MPC, CY-MPC, and MPC-LACBWR ISFSI storage pad layouts are shown in Figures 1.4-1, 1.4-2, and 1.A.4-1, respectively, of the NAC-MPC System FSAR [2.7.1.a thru 2.7.1.m]. The final ISFSI pad designs have significant differences from the FSAR conceptual figures. The ISFSI storage pad is a steel-reinforced concrete slab that supports free-standing NAC-MPC System casks. As discussed in Section 1.4 of the NAC-MPC System FSAR, the ISFSI storage pad can support the loads from the NAC-MPC System casks. Some NAC-MPC System users have identified the ISFSI storage pad as ITS (Category C) components and will perform aging management inspections on a site-specific basis independent of the CoC renewal.

2.4.9 ISFSI Security Equipment

The ISFSI security equipment (e.g., ISFSI security fences and gates, lighting, communications, monitoring equipment, etc.) are not part of the NAC-MPC System approved by the NAC-MPC System CoC [2.7.2.a thru 2.7.2.i], and as such, are not described in the NAC-MPC System FSAR [2.7.1.a thru 2.7.1.m]. Existing plant programs and procedures ensure that the ISFSI security equipment requirements are met in accordance with 10 CFR 73. Furthermore, potential failure of the ISFSI security equipment would not prevent the NAC-MPC System casks from performing their intended functions. NUREG-1927 specifically excludes inclusion of ISFSI security equipment in the application for recertification for a period of extended operation.

2.5 SSC WITHIN SCOPE OF CoC RENEWAL APPLICATION

The SSCs determined to be within the scope of renewal are the TSC, VCC, Transfer Cask (TFR)/Transfer Adapter, and the loaded spent nuclear fuel (SNF) assemblies. These basic components are the only SSC ITS approved by the CoC [2.7.2.a thru 2.7.2.i] under 10 CFR 72, Subpart L. The TSC, VCC, TFR/Transfer Adapter, and SNF all satisfy Criterion 1 of the scoping evaluation.

The intended functions performed by the individual subcomponents of the in-scope SSCs are identified in the summary tables for the TSC and Fuel Basket, Vertical Concrete Cask, Transfer Cask/Transfer Adapter and Spent Fuel Assemblies, Tables 2.5-1 thru 2.5-9. The important safety functions are defined by the following:

- Thermal/Heat Removal (TH)
- Structural Integrity (SR)
- Confinement (CO)
- Radiation Shielding (SH)
- Subcriticality (CR)
- Retrievability (RE)

The applicable license drawings were reviewed to identify the SSC subcomponents that are ITS in accordance with criterion 1 of the scoping process. Following the initial review, SSC subcomponents identified as NITS were reviewed under the scoping process criterion 2, which identifies subcomponents whose failure could impact the performance of ITS SSC subcomponents. The criterion 2 review identified additional SSC subcomponents that will require evaluation as in scope for the CoC renewal evaluations and are so identified on the SSC subcomponent tables.

2.6 SSC NOT WITHIN SCOPE OF CoC RENEWAL APPLICATION

The SSC that are not in the scope of NAC-MPC System CoC renewal include fuel transfer and auxiliary equipment, temperature monitoring systems, ISFSI storage pad, and ISFSI security equipment. These components are classified as NITS and do not meet scoping criterion 2 except for ISFSI storage pad which requires aging management by the General Licensee, if identified as an ITS Category C component on a site-specific basis.

2.6.1 Fuel Transfer and Auxiliary Equipment

The fuel transfer and auxiliary equipment necessary for ISFSI operations (e.g., lifting yoke, air-pallets, heavy haul trailer, vertical cask transporter, vacuum drying system, welding equipment, weld inspection equipment, drain pump equipment, temperature monitoring equipment, and helium leak detection equipment, etc.) are not included as part of the NAC-MPC System certified by the NRC in the NAC-MPC System CoC No. 1025 [2.7.2.a thru 2.7.2.i] and as such, are not described in detail in the NAC-MPC System FSARs [2.7.1.a thru 2.7.1.m]. The failure of the fuel transfer and auxiliary equipment would not prevent the TSC, VCC, TFR, or SFAs from fulfilling their intended safety functions. Therefore, the fuel transfer and auxiliary equipment do not meet scoping criterion 2 and are not within the scope of the CoC renewal. The fuel transfer and auxiliary equipment are addressed in site-specific reviews. A majority of this equipment was disposed of following completion of the spent fuel loading operations and decommissioning of the reactor plant. When required for de-inventory operations for removing the loaded NAC-MPC TSCs from the ISFSIs, new or refurbished equipment will be provided to complete the fuel transfer operations.

2.6.2 VCC Temperature Monitoring System

The NAC-MPC System VCC temperature monitoring system is one method authorized to verify the continued operability of the VCC heat removal system, although it is not part of the system authorized by the NRC in the NAC-MPC System CoC No. 1025 [2.7.2.a thru 2.7.2.i], and as such, is not described in detail in the NAC-MPC System FSARs [2.7.1.a thru 2.7.1.m]. Typically, a VCC temperature monitoring system is provided by thermocouples or RTDs placed in each of the four outlet vents. The average outlet temperature is compared to the ISFSI pad ambient temperature to verify the temperature differential is below the Technical Specification allowable every 24 hours. Alternatively, a visual inspection may be performed on a 24-hour frequency to verify that the inlet and outlet screens are unobstructed. The failure of the temperature monitoring equipment would not prevent the VCC from maintaining the stored fuel cladding and MPC components within allowable temperature limits for a period exceeding 24 hours to allow corrective actions to be taken to re-establish operability of the VCC heat removal system. Therefore, the VCC temperature monitoring system does not meet scoping criterion 2 and are not within the scope of the CoC renewal.

2.6.3 ISFSI Storage Pad

The NAC-MPC System ISFSI storage pad is not part of the NAC-MPC System certified by the NRC in the NAC-MPC CoC No. 1025 [2.7.2.a thru 2.7.2.i] under 10 CFR Part 72, Subpart L. The ISFSI storage pad provides free-standing support of the NAC-MPC System casks. The generic requirements for the ISFSI physical parameters are addressed in the USFARs [2.7.1.a thru 2.7.1.m] in the evaluation of VCC accident drops and the beyond design basis tip-over accident. The FSAR and CoC authorize the evaluation of the ISFSI pad on a site-specific basis as part of the 10 CFR 72.212 evaluation. However, the ISFSI storage pad meets scoping criterion 1 if the pad is classified as ITS Category C by the General Licensee. Although not within the scope of NAC-MPC System CoC renewal, the aging management inspections, if required, of the ISFSI pad will be addressed on a site-specific inspection program basis by the General Licensee.

2.6.4 ISFSI Security Equipment

The ISFSI security equipment is not within the scope of CoC renewal per NUREG-1927 Rev 1.

Table 2.2-1 Applicable YR-MPC License Drawings - (Revision Number and Number of Sheets Indicated)

Drawing Number	Drawing Title	FSAR R0(1)	FSAR R1(1)	FSAR R2(1)	FSAR R3(1)	FSAR R4(1)	FSAR R5(1)	FSAR R6(1)	FSAR R7(1)	FSAR R8(1)	FSAR R9(1)	FSAR R10(1)	FSAR R11(1)	FSAR R12(1)
455-821	STC Adapter Ring	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1
455-856	VCC Nameplate	0/1	1/1	1/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1
455-859	Transfer Adapter	1/3	3/3	5/4	5/4	5/4	5/4	5/4	5/4	5/4	6/4	6/4	6/4	6/4
455-860	Transfer Cask	4/4	6/5	8/5	10/5	10/5	10/5	10/5	10/5	10/5	11/5	11/5	11/5	11/5
455-861	VCC Structural Weldments	4/2	6/3	7/3	7/3	7/3	7/3	7/3	8/3	8/3	8/3	8/3	8/3	8/3
455-862	Loaded VCC	2/1	3/1	6/2	7/2	7/2	7/2	7/2	8/2	8/2	8/2	8/2	9/2	9/2
455-863	VCC Lid	2/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1
455-864	VCC Shield Plug	1/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1
455-866	VCC Reinforcing Bar and Concrete	0/3	4/4	4/4	5/4	5/4	5/4	5/4	5/4	5/4	5/4	5/4	5/4	5/4
455-870	Canister Shell	3/1	4/1	4/1	5/1	5/1	5/1	5/1	5/1	5/1	5/1	5/1	5/1	5/1
455-871	Canister Details	4/2	6/2	7/2	8/2	8/2	8/2	8/2	8/2	8/2	8/2	8/2	8/2	8/2
455-871	Canister Details	-	-	-	7P2/3	7P2/3	7P2/3	7P2/3	7P2/3	7P2/3	7P2/3	7P2/3	7P2/3	7P2/3
455-872	TSC Assembly	6/2	9/2	11/2	12/2	12/2	12/2	12/2	12/2	12/2	12/2	12/2	12/2	12/2
455-872	TSC Assembly	-	-	-	11P1/2	11P1/2	11P1/2	11P1/2	11P1/2	11P1/2	11P1/2	11P1/2	11P1/2	11P1/2
455-873	Drain Tube Assy.	2/1	3/1	3/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1
455-881	PWR Fuel Tube	3/1	7/3	8/3	8/3	8/3	8/3	8/3	8/3	8/3	8/3	8/3	8/3	8/3
455-891	Fuel Basket (FB) Bottom Weldment	0/1	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2
455-891	FB Bottom Weldment	-	-	-	2P0/3	2P0/3	2P0/3	2P0/3	2P0/3	2P0/3	2P0/3	2P0/3	2P0/3	2P0/3

Table 2.2-1 Applicable YR-MPC License Drawings - (Revision Number and Number of Sheets Indicated)

Drawing Number	Drawing Title	FSAR R0[(1)]	FSAR R1[(1)]	FSAR R2[(1)]	FSAR R3[(1)]	FSAR R4[(1)]	FSAR R5[(1)]	FSAR R6[(1)]	FSAR R7[(1)]	FSAR R8[(1)]	FSAR R9[(1)]	FSAR R10[(1)]	FSAR R11[(1)]	FSAR R12[(1)]
455-892	FB Top Weldment	1/1	2/2	3/2	3/2	3/2	3/2	3/2	3/2	3/2	3/2	3/2	3/2	3/2
455-892	FB Top Weldment	-	-		3P0/3	3P0/3	3P0/3	3P0/3	3P0/3	3P0/3	3P0/3	3P0/3	3P0/3	3P0/3
455-893	FB Support Disk	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1
455-894	FB Heat Transfer Disk	1/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1
455-895	FB Assembly	2/1	4/2	4/2	5/2	5/2	5/2	5/2	5/2	5/2	5/2	5/2	5/2	5/2
455-895	FB Assembly	-	-	-	5P0/2	5P0/2	5P0/2	5P0/2	5P0/2	5P0/2	5P0/2	5P0/2	5P0/2	5P0/2
455-901	DFC Assembly	-	-	-	0P0/2	0P0/2	0P0/2	0P0/2	0P0/2	0P0/2	0P0/2	0P0/2	0P0/2	0P0/2
455-902	DFC Details	-	-	-	0P4/5	0P4/5	0P4/5	0P4/5	0P4/5	0P4/5	0P4/5	0P4/5	0P4/5	0P4/5
455-913	VCC Supplemental Shielding	-	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1
455-918	TFR Door Stop	-	0/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2
455-919	United Nuclear Test Assembly Retainer	-	-	0/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1
YR-00-060	Yankee Class Reconfigured Fuel Assembly (RFA)	1/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1
YR-00-061	RFA Shell Weldment	1/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1
YR-00-062 Sheet 1	RFA Top End Fitting	1/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1
YR-00-062 Sheet 2	RFA Top End Fitting	-	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1

Table 2.2-1 Applicable YR-MPC License Drawings - (Revision Number and Number of Sheets Indicated)

Drawing Number	Drawing Title	FSAR R0(1)	FSAR R1(1)	FSAR R2(1)	FSAR R3(1)	FSAR R4(1)	FSAR R5(1)	FSAR R6(1)	FSAR R7(1)	FSAR R8(1)	FSAR R9(1)	FSAR R10(1)	FSAR R11(1)	FSAR R12(1)
YR-00-062 Sheet 3	RFA Top End Fitting	-	D1/1	D1/1	D1/1	D1/1	D1/1	D1/1	D1/1	D1/1	D1/1	D1/1	D1/1	D1/1
YR-00-063	RFA Bottom End Fitting	1/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1
YR-00-064	RFA Nozzle Bolt	1/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1	D4/1
YR-00-065	RFA Fuel Basket	1/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1	D2/1
YR-00-066 Sheet 1	RFA Fuel Tube	1/1	D5/1	D5/1	D5/1	D5/1	D5/1	D5/1	D5/1	D5/1	D5/1	D5/1	D5/1	D5/1
YR-00-066 Sheet 2	RFA Fuel Tube	-	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1	D3/1

Note:

(1) NAC-MPC System Final Safety Analysis Report and applicable revision number. The revision of the drawing and number of sheets are indicated for each drawing listed.

Table 2.2-2 Applicable CY-MPC License Drawings - (Revision Number and Number of Sheets Indicated)

Drawing Number	Drawing Title	FSAR R2(1)(2)	FSAR R3(1)	FSAR R4(1)	FSAR R5(1)	FSAR R6(1)	FSAR R7(1)	FSAR R8(1)	FSAR R9(1)	FSAR R10(1)	FSAR R11(1)	FSAR R12(1)
455-821	STC Adapter Ring	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1
414-856	VCC Nameplate	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1
455-859	Transfer Adapter	5/4	5/4	5/4	5/4	5/4	5/4	5/4	5/4	5/4	5/4	5/4
414-860	Transfer Cask	4/5	4/5	5/5	6/5	6/5	6/5	6/5	6/5	6/5	6/5	6/5
414-861	VCC Structural Weldments	7/3	7/3	7/3	8/3	8/3	8/3	8/3	8/3	8/3	8/3	8/3
414-862	Loaded VCC	4/1	4/1	4/1	4/1	4/1	5/1	5/1	5/1	5/1	6/2	6/2
414-863	VCC Lid	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1
414-864	VCC Shield Plug	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1
414-866	VCC Reinforcing Bar and Concrete	4/4	4/4	4/4	4/4	5/6	5/6	5/6	5/6	5/6	5/6	5/6
414-870	Canister Shell	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1
414-871	Canister Details	3/2	5/2	6/2	6/2	6/2	6/2	6/2	6/2	6/2	6/2	6/2
414-872	TSC Assembly	3/3	5/3	6/3	6/3	6/3	6/3	6/3	6/3	6/3	6/3	6/3
414-873	Drain Tube Assy.	0/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1
414-891	Fuel Basket (FB) Bottom Weldment	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1
414-894	FB Heat Transfer Disk	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1	0/1
414-895	FB Assembly	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2
414-901	DFC Assembly	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1
414-902	DFC Details	2/3	3/3	3/3	3/3	3/3	3/3	3/3	3/3	3/3	3/3	3/3
414-903	Reconfigured Fuel Assembly (RFA)	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2
414-904	RFA Details	0/3	0/3	0/3	0/3	0/3	0/3	0/3	0/3	0/3	0/3	0/3
414-917	TFR Door Stop	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2	1/2

Note:

(1) NAC-MPC System Final Safety Analysis Report and applicable revision number. The revision of the drawing and number of sheets are indicated for each drawing listed.

(2) First revision with CY-MPC specific License Drawings.

Table 2.2-3 Applicable MPC-LACBWR License Drawings - (Revision Number and Number of Sheets Indicated)

Drawing Number	Drawing Title	FSAR R8(1)(2)	FSAR R9(1)	FSAR R10(1)	FSAR R11(1)	FSAR R12(1)
455-859	Transfer Adapter Assy.	5/4	6/4	6/4	6/4	6/4
455-860	Transfer Cask Assembly	10/5	11/5	11/5	11/5	11/5
630045-861	VCC Structural Weldment	1/3	3/3	4/3	4/3	4/3
630045-862	Loaded VCC	0/1	0/1	0/1	2/1	2/1
630045-863	VCC Lid Assembly	0/1	1/1	2/1	2/1	2/1
630045-864	VCC Nameplate	0/1	2/1	2/1	2/1	2/1
630045-866	VCC Reinforcing Bar and Concrete Placement	1/5	1/5	1/5	1/5	4/7
630045-870	Canister Shell Weldment	0/1	3/1	3/1	3/1	3/1
630045-871	TSC Details	0/4	5/4	5/4	5/4	5/4
630045-872	TSC Assembly	0/2	5/2	6/2	6/2	6/2
630045-873	TSC Drain Tube Assembly	0/1	1/1	1/1	1/1	1/1
630045-877	Fuel Basket (FB) Bottom Weldment	0/1	3/1	3/1	3/1	3/1
630045-878	FB Top Weldment	0/1	1/1	1/1	1/1	1/1
630045-881	Fuel Tube Assembly	0/2	1/2	1/2	1/2	1/2
630045-893	FB Support Disk	0/1	1/1	1/1	1/1	1/1
630045-894	FB Heat Transfer Disk	0/1	1/1	1/1	1/1	1/1
630045-895	Fuel Basket Assembly – 68 BWR	0/3	2/3	2/3	2/3	2/3
630045-901	DFC Assembly	0/1	0/1	0/1	0/1	0/1
630045-902	DFC Details	0/2	1/2	1/2	1/2	1/2

Note:

(1) NAC-MPC System Final Safety Analysis Report and applicable revision number. The revision of the drawing and number of sheets are indicated for each drawing listed.

(2) First revision with MPC-LACBWR specific License Drawings

Table 2.3-1 Summary of Scoping Evaluation Results for NAC-MPC Systems

SSC Description	Scoping Results		In-Scope SSC
	Criterion 1 (1)	Criterion 2 (2)	
Transportable Storage Canister (TSC/Canister)	Yes	NA	Yes
Vertical Concrete Cask (VCC)	Yes	NA	Yes
Transfer Cask (TFR)	Yes	NA	Yes (7)
Transfer Adapter Plate	Yes	NA	Yes (7)
Spent Nuclear Fuel Assemblies	Yes	NA	Yes (3)
Fuel Transfer Equipment (4) and Ancillary Operating Equipment (5)	No	No	No
Temperature Monitoring Equipment	No	No	No
ISFSI Storage Pad (8)	Yes (8)	No (9)	Yes (8)
ISFSI Security Equipment (6)	No	No	No

Notes:

(1) SSC is Important-to-Safety (ITS).
(2) SSC is Not-Important-to-Safety (NITS), but its failure could prevent an ITS function from being fulfilled.
(3) Fuel pellets are not within the scope of the renewal.
(4) Fuel transfer equipment includes a) hardware to position the transfer cask with respect to the storage or transport cask; b) lifting yoke for the transfer cask; c) lifting slings for the canister and canister lids, d) air pallets, e) heavy haul trailer, and f) vertical cask transporter (applicable to facilities that still retain transfer equipment on site).
(5) Ancillary equipment includes canister closure equipment used to drain, backfill, and seal the canister (e.g., the suction pump equipment, the vacuum drying system, automated or manual welding equipment, weld inspection equipment, helium backfill and leak detection equipment, etc.).
(6) ISFSI security equipment includes the ISFSI security fences and gates, lighting, communications, and monitoring equipment is specifically excluded from the scope of CoC renewal per NUREG-1927 Rev 1.
(7) Applicable to sites that still retain a Transfer Cask (TFR) and/or Transfer Adapter Plate on-site, and to TFRs in storage under NAC control. NA to facilities that have disposed of the equipment, or the equipment is no longer available. Sites requiring TFR and Transfer Adapters for final transfer of the TSC into a transport cask will be provided with and use the required equipment meeting all applicable Aging Management Program requirements.
(8) ISFSI storage pads identified by General Licensees as being ITS Category C shall have aging management implemented by the General Licensee outside scope of CoC Renewal.
(9) ISFSI storage pad if designated as NITS by the General Licensee.

Table 2.5-1 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents for YR-MPC

Subcomponent	Part or I.D. No.	Reference Drawing [(1)]	Intended Safety Function(s) [(2)]	Quality Category	Sub-Scoping Results		In-Scope [(3)]
					Criterion 1	Criterion 2	
TSC Shell	Item 1	455-870	SR, CO, RE	A	X	---	Yes
Bottom	Item 2	455-870	SR, CO, RE	A	X	---	Yes
Location Lug	Item 3	455-870	---	B	---	---	No
Weather Resistant Paint (Alignment Mark) on TSC Shell	Dwg. Note 2	455-870	---	NQ	---	---	No
Shield Lid Support Ring	Item 1	455-871	SR, SH, RE	B	X	---	Yes
Spacer Ring	Item 2	455-871	SR	B	X	---	Yes
Shield Lid	Item 3	455-871	SR, CO, SH	B	X	---	Yes
Metal Boss Seal	Item 4	455-871	---	C	---	---	No
Structural Lid	Item 5	455-871	SR, RE	A	X	---	Yes
Valved Nipple	Item 6	455-871	---	C	---	---	No
Port Cover	Item 7	455-871	CO	B	X	---	Yes
Key	Item 8	455-871	---	C	---	---	No
Shield Lid – Damaged Fuel	Item 9	455-871-7-P2	SR, CO, SH	B	X	---	Yes
Weather Resistant Paint (Alignment Mark) on Structural Lid	Dwg. Note 2	455-871	---	NQ	---	---	No
Shield Lid Plug	Item 10	455-872	---	NQ	---	---	No
Structural Lid Plug	Item 11	455-872	---	NQ	---	---	No
Dowel Pin	Item 12	455-872	---	NQ	---	---	No
Valved Nipple	Item 1	455-873	---	C	---	---	No
Tube	Item 2	455-873	---	C	---	---	No
Metal Boss Seal	Item 3	455-873	---	C	---	---	No
PWR Fuel Tube	Items 1 & 5	455-881	CR	A	X	---	Yes
Neutron Absorber	Item 2	455-881	CR	A	X	---	Yes

Table 2.5-1 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents for YR-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Cladding	Item 3	455-881	SR, CR	A	X	---	Yes
Tube Flange	Item 4 & 6	455-881	SR	A	X	---	Yes
Bottom Fuel Basket (FB) Plate	Item 1	455-891	SR	A	X	---	Yes
Bottom FB Weldment Pad	Item 2	455-891	SR	A	X	---	Yes
Bottom FB Weldment Support Plate	Items 3-4	455-891	SR	A	X	---	Yes
Bottom Oversized FB Plate	Item 5	455-891	SR	A	X	---	Yes
Bottom Weldment FB Plate – Damaged Fuel	Item 6	455-891	SR	A	X	---	Yes
Top FB Plate	Item 1	455-892	SR	A	X	---	Yes
Top FB Structural Ring	Item 2	455-892	SR	A	X	---	Yes
Top FB Weldment Support Plate	Items 3	455-892	SR	A	X	---	Yes
Top FB Oversized Plate	Item 4	455-892	SR	A	X	---	Yes
Top FB Plate – Damaged Fuel	Item 5	455-892	SR	A	X	---	Yes
FB Support Disk	Item 1	455-893	SR	A	X	---	Yes
Spacer	Item 2	455-893	SR	A	X	---	Yes
Bottom Spacer	Item 3	455-893	SR	A	X	---	Yes
Top Nut	Item 4	455-893	SR	A	X	---	Yes
Tie Rod	Item 5	455-893	SR	A	X	---	Yes
Split Spacer	Item 6	455-893	SR	A	X	---	Yes
Top Spacer	Item 7	455-893	SR	A	X	---	Yes
FB Heat Transfer Disk	Item 1	455-894	TH	A	X	---	Yes
PWR Drain Tube Sleeve	Item 4	455-895	---	C	---	---	No

Table 2.5-1 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents for YR-MPC

Subcomponent	Part or I.D. No.	Reference Drawing [(1)]	Intended Safety Function(s) [(2)]	Quality Category	Sub-Scoping Results		In-Scope [(3)]
					Criterion 1	Criterion 2	
PWR Basket Flat Washer	Item 13	455-895	SR	B	X	---	Yes
Top Weldment Baffle A	Item 16	455-895	SR	A	X	---	Yes
Top Weldment Baffle B	Item 17	455-895	SR	A	X	---	Yes
Screen Cover Plate	Item 1	455-902	SR	A	X	---	Yes
Damaged Fuel Can (DFC) Lid Plate	Item 2	455-902	SR, CR	A	X	---	Yes
Lid Guide	Item 3	455-902	---	C	---	---	No
Wiper	Item 4	455-902	---	C	X	---	Yes
Lid Bottom Plate	Item 5	455-902	SR, CR	A	X	---	Yes
Filter Screen	Items 6 & 14	455-902	CO	C	X	---	Yes
Backing Screen	Items 7 & 15	455-902	CO	C	X	---	Yes
DFC Bottom Plate	Item 8	455-902	SR, CR	A	X	---	Yes
DFC Collar Side Plate	Item 9	455-902	SR	A	X	---	Yes
DFC Tube Body	Item 10	455-902	CR	A	X	---	Yes
Lift Tee	Item 12	455-902	SR	B	X	---	Yes
Support Ring	Item 13	455-902	SR	B	X	---	Yes
Dowel Pin	Item 16	455-902	SR	C	X	---	Yes
Test Assembly Retainer Lower Tab	Item 1	455-919	SR	A	X	---	Yes
Sleeve	Item 2	455-919	SR	A	X	---	Yes
Lifting Plate	Item 3	455-919	SR	A	X	---	Yes
Gusset	Item 4	455-919	SR	A	X	---	Yes
Ring	Item 5	455-919	SR	A	X	---	Yes
RFA Shell Casing	Item 1	YR-00-061	SR, CR	A	X	---	Yes
RFA Top Ring	Item 2	YR-00-061	SR, CR	A	X	---	Yes

Table 2.5-1 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents for YR-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
RFA Top End Fitting	Item 1	YR-00-062, Sh. 1	SR, CR	A	X	---	Yes
RFA Cylinder	Item 2	YR-00-062, Sh. 1	SR, CR	A	X	---	Yes
RFA Inside Tab	Item 3	YR-00-062, Sh. 1	SR, CR	A	X	---	Yes
RFA Outside Tab	Item 4	YR-00-062, Sh. 1	SR, CR	A	X	---	Yes
RFA Top End Plate	Item 1	YR-00-062, Sh. 2	SR, CR	A	X	---	Yes
RFA Top End Template	Item 10	YR-00-062, Sh. 3	SR, CR	A	X	---	Yes
RFA Bottom End Fitting	Items 1-5	YR-00-063	SR, CR	A	X	---	Yes
RFA Bolt	Item 1	YR-00-064	SR, CR	A	X	---	Yes
RFA Alignment Pin	Item 5	YR-00-064	SR, CR	C	X	---	Yes
RFA Fuel Basket Corner Angle	Item 1	YR-00-065	SR, CR	A	X	---	Yes
RFA Fuel Basket Tie Plate	Item 2	YR-00-065	SR, CR	A	X	---	Yes
RFA Fuel Basket Fuel Tube	Item 1	YR-00-066	SR, CR	A	X	---	Yes
RFA Fuel Basket Top Cap	Item 2	YR-00-066	SR, CR	A	X	---	Yes
RFA Fuel Basket Bottom Cap	Item 3	YR-00-066	SR, CR	A	X	---	Yes

Notes:

(1) Included in Section 1.7 of the NAC-MPC System Updated Final Safety Analysis Report (FSAR) [2.7.1.a – 2.7.1.m]

(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Subcriticality (CR), and Retrievability (RE)

(3) Items identified as No in the In-Scope column do not have an identified ITS function and do not require aging management review.

(4) Non-Quality (NQ) is used for NITS designation.

Table 2.5-2 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents for CY-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
TSC Shell	Item 1	414-870	SR, CO, RE	A	X	---	Yes
Bottom	Item 2	414-870	SR, CO, RE	A	X	---	Yes
Location Lug	Item 3	414-870	---	C	---	---	No
Paint	Item 4	414-870	---	NQ	---	---	No
Weather Resistant Paint (Alignment Mark)	Dwg. Note 2	414-870	---	NQ	---	---	No
Shield Lid Support Ring	Item 1	414-871	SR, SH, RE	B	X	---	Yes
Spacer Ring	Item 2	414-871	SR	C	X	---	Yes
Shield Lid	Item 3	414-871	SR, CO, SH	B	X	---	Yes
Key	Item 4	414-871	---	C	---	---	No
Structural Lid	Item 5	414-871	SR, RE	A	X	---	Yes
Valved Nipple	Item 6	414-871	---	C	---	---	No
Port Cover	Item 7	414-871	CO	B	X	---	Yes
Seal	Item 8	414-871	---	C	---	---	No
Lubricant	Item 9	414-871	---	NQ	---	---	No
Weather Resistant Paint (Alignment Mark) on Structural Lid	Dwg. Note 2	414-871	---	NQ	---	---	No
Shield Lid Plug	Item 10	414-872	---	NQ	---	---	No
Structural Lid Plug	Item 11	414-872	---	NQ	---	---	No
Dowel Pin	Item 13	414-872	---	NQ	---	---	No
Lubricant	Item 14	414-872	---	NQ	---	---	No
Valved Nipple	Item 1	414-873	---	C	---	---	No
Tube	Item 2	414-873	---	C	---	---	No
Seal	Item 3	414-873	---	C	---	---	No

Table 2.5-2 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents for CY-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
PWR Fuel Tube	Item 1	414-881	CR	A	X	---	Yes
Neutron Absorber	Item 2	414-881	CR	A	X	---	Yes
Cladding	Item 3	414-881	SR, CR	A	X	---	Yes
Tube Flange	Item 4	414-881	SR	A	X	---	Yes
PWR Oversized Fuel Tube	Item 1	414-882	CR	A	X	---	Yes
Neutron Absorber	Item 2	414-882	CR	A	X	---	Yes
Cladding	Item 3	414-882	SR, CR	A	X	---	Yes
Tube Flange	Item 4	414-882	SR	A	X	---	Yes
Bottom Fuel Basket (FB) Plate	Item 1	414-891	SR	A	X	---	Yes
Bottom FB Weldment Pad	Item 2	414-891	SR	A	X	---	Yes
Bottom FB Weldment Support Plate	Items 3-6	414-891	SR	A	X	---	Yes
Top FB (4) Plate	Items 1 & 6	414-892	SR	A	X	---	Yes
Top FB Structural Ring	Item 2	414-892	SR	A	X	---	Yes
Top FB Weldment Support Plate	Item 3	414-892	SR	A	X	---	Yes
Baffle	Items 4 & 5	414-892	SR	C	X	---	Yes
FB Shield Baffle	Item 7	414-892	SR	C	X	---	Yes
FB Support Disk	Item 1	414-893	SR	A	X	---	Yes
FB Bottom Spacer	Item 2	414-893	SR	A	X	---	Yes
Top Spacer	Item 3	414-893	SR	A	X	---	Yes
Top Nut	Item 4	414-893	SR	A	X	---	Yes
Tie Rod	Item 5	414-893	SR	A	X	---	Yes
Split Spacer	Item 6	414-893	SR	A	X	---	Yes

Table 2.5-2 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents for CY-MPC

Subcomponent	Part or I.D. No.	Reference Drawing [(1)]	Intended Safety Function(s) [(2)]	Quality Category	Sub-Scoping Results		In-Scope [(3)]
					Criterion 1	Criterion 2	
Washer	Item 7	414-893	SR	C	X	---	Yes
FB Heat Transfer Disk	Item 1	414-894	TH	B	X	---	Yes
PWR Drain Tube Sleeve	Item 4	414-895	---	C	---	---	No
Lubricant	Item 15	414-895	---	NQ	---	---	No
Damaged Fuel Can (DFC) Collar	Item 1	414-902	---	C	X	---	Yes
DFC Lid Plate	Item 2	414-902	SR	A	X	---	Yes
Lid Guide	Item 3	414-902	---	C	---	---	No
Wiper	Item 4	414-902	CO	C	X	---	Yes
DFC Bottom Plate	Item 5	414-902	SR, CO	A	X	---	Yes
Filter Screen	Items 6 & 14	414-902	CO	C	X	---	Yes
Backing Screen	Items 7 & 15	414-902	CO	C	X	---	Yes
Side Plate	Item 8	414-902	SR, CR	A	X	---	Yes
DFC Tube Body	Item 9	414-902	CR	A	X	---	Yes
Lift Tee	Item 11	414-902	SR	B	X	---	Yes
Support Ring	Item 12	414-902	SR	B	X	---	Yes
Lid Bottom Plate	Item 13	414-902	SR	A	X	---	Yes
Dowel Pin	Item 16	414-902	SR	C	X	---	Yes
RFA Corner Angle	Item 4	414-903	SR, CR	A	X	---	Yes
RFA Tube	Item 5	414-903	SR, CR	A	X	---	Yes
Filter Screen	Item 8	414-903	---	C	X	---	Yes
Backing Screen	Item 9	414-903	---	C	X	---	Yes
Stand-off Pin	Item 10	414-903	SR	C	X	---	Yes
Hex Head Bolt	Item 16	414-903	SR	C	X	---	Yes
Support Grid	Item 17	414-903	SR, CR	B	X	---	Yes
RFA Bottom Housing	Item 1	414-904	SR, CR	A	X	---	Yes
Retaining Plate	Item 2	414-904	SR, CR	A	X	---	Yes

Table 2.5-2 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents for CY-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Retaining Ring	Item 3	414-904	SR	A	X	---	Yes
RFA Top Housing	Item 4	414-904	SR	A	X	---	Yes
Guide Plate	Item 5	414-904	SR	C	X	---	Yes
Rod Retaining Plate	Item 6	414-904	SR	A	X	---	Yes
Screen Ring	Item 7	414-904	SR	A	X	---	Yes
Screen Housing	Item 8	414-904	SR	A	X	---	Yes

Notes:

(1) Included in Section 1.7 of the NAC-MPC System Updated Final Safety Analysis Report (FSAR) [2.7.1.a – 2.7.1.m]

(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Subcriticality (CR), and Retrievability (RE)

(3) Items identified as No in the In-Scope column do not have an identified ITS function and do not require aging management review.

(4) Non-Quality (NQ) is used for NITS designation.

Table 2.5-3 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents MPC-LACBWR

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
TSC Shell	Item 1	630045-870	SR, CO, RE	A	X	---	Yes
Bottom Plate	Item 2	630045-870	SR, CO, RE	A	X	---	Yes
Location Lug	Item 3	630045-870	---	C	---	---	No
Weather Resistant Paint (Alignment Mark) on TSC Shell	Dwg. Note 9	630045-870	---	NQ	---	---	No
Closure Lid	Item 1	630045-871	SR, CO, RE	A	X	---	Yes
Nipple	Item 2	630045-871	---	NQ	---	---	No
Seal	Item 3	630045-871	---	NQ	---	---	No
Closure Lid Support Ring	Item 4	630045-871	SR	A	X	---	Yes
Inner Port Cover	Item 5	630045-871	SR, CO	A	X	---	Yes
Key	Item 6	630045-871	---	C	---	---	No
Closure Ring	Item 7	630045-871	SR, CO, RE	A	X	---	Yes
Closure Lid Plug	Item 8	630045-871	---	NQ	---	---	No
Spacer	Item 9	630045-871	SR, CO	B	X	---	Yes
Bolt	Item 10	630045-871	SR	B	X	---	Yes
Nord-Lock Washer	Item 11	630045-871	SR	C	X	---	Yes
Outer Port Cover	Item 12	630045-871	SR, CO	A	X	---	Yes
Weather Resistant Paint (Alignment Mark) on Closure Lid	Dwg. Note 1	630045-871	---	NQ	---	---	No
Drain Tube Nipple	Item 1	630045-873	---	NQ	---	---	No
Drain Tube	Item 2	630045-873	---	NQ	---	---	No
Seal	Item 3	630045-873	---	NQ	---	---	No
Bottom Fuel Basket (FB) Plate	Item 1	630045-877	SR	A	X	---	Yes
Bottom FB Weldment Pad	Item 2	630045-877	SR	A	X	---	Yes

Table 2.5-3 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents MPC-LACBWR

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Bottom FB Weldment Support Plate	Item 3	630045-877	SR	A	X	---	Yes
Top FB Weldment Plate	Item 1	630045-878	SR	A	X	---	Yes
Top FB Weldment Ring	Item 2	630045-878	SR	A	X	---	Yes
Top FB Weldment Support Plate	Items 3-5 & 8	630045-878	SR	A	X	---	Yes
Top FB Weldment Stiffener-A	Item 6	630045-878	SR	A	X	---	Yes
Top FB Weldment Stiffener-B	Item 7	630045-878	SR	A	X	---	Yes
BWR Fuel Tube	Item 1	630045-881	CR	A	X	---	Yes
Neutron Absorber	Items 2 & 6	630045-881	CR	A	X	---	Yes
Cladding	Items 3 & 7	630045-881	SR, CR	A	X	---	Yes
Tube Flange	Item 4	630045-881	SR	A	X	---	Yes
Plate	Item 5	630045-881	TH	A	X	---	Yes
FB Support Disk	Items 1 - 3	630045-893	SR	A	X	---	Yes
FB Heat Transfer Disk	Item 1	630045-894	TH	A	X	---	Yes
Drain Tube Sleeve(3)	Item 4	630045-895	---	NQ	---	---	No
Spacer	Items 7, 21 & 22	630045-895	SR	A	X	---	Yes
Bottom Spacer	Item 8	630045-895	SR	A	X	---	Yes
Top Nut	Item 10	630045-895	SR	A	X	---	Yes
Tie Rods	Items 11	630045-895	SR	A	X	---	Yes
Top Spacer	Item 12	630045-895	SR	A	X	---	Yes
Split Spacer	Item 13	630045-895	SR	A	X	---	Yes
Flat Washer	Item 14	630045-895	SR	C	X	---	Yes

Table 2.5-3 Intended Functions of NAC-MPC Transportable Storage Canister (TSC) Subcomponents MPC-LACBWR

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Damaged Fuel Can (DFC) Collar	Item 1	630045-902	SR	A	X	---	Yes
DFC Lid Plate	Item 2	630045-902	SR, CR	A	X	---	Yes
Lid Guide	Item 3	630045-902	---	C	---	---	No
Wiper	Item 4	630045-902	---	C	X	---	Yes
DFC Bottom Plate	Item 5	630045-902	SR, CR	A	X	---	Yes
Filter Screen	Items 6 & 14	630045-902	CO	C	X	---	Yes
Backing Screen	Items 7 & 15	630045-902	CO	C	X	---	Yes
Side Plate	Item 8	630045-902	SR, CR	A	X	---	Yes
DFC Tube Body	Item 9	630045-902	CR	A	X	---	Yes
Lift Tee	Item 11	630045-902	SR	B	X	---	Yes
Support Ring	Item 12	630045-902	SR	B	X	---	Yes
Lid Bottom Plate	Item 13	630045-902	SR, CR	A	X	---	Yes
Dowel Pin	Item 16	630045-902	SR	C	X	---	Yes

Notes:

(1) Included in Section 1.A.7 of the NAC-MPC System Updated Final Safety Analysis Report (FSAR) [2.7.1.a – 2.7.1.m]
(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Subcriticality (CR), and Retrievability (RE)
(3) Items identified as No in the In-Scope column do not have an identified ITS function and do not require aging management review.
(4) Non-Quality (NQ) is used for NITS designation.

Table 2.5-4 Intended Functions of NAC-MPC Vertical Concrete Cask (VCC) Subcomponents YR-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
VCC Liner Shell	Item 1	455-861	SH, TH, SR	B	X	---	Yes
Top Flange	Item 2	455-861	SR	B	X	---	Yes
Support Ring	Item 3	455-861	SR	B	X	---	Yes
Jack Base	Item 4	455-861	---	C	---	---	No
Jack Gusset	Item 5	455-861	---	C	---	---	No
Jack Screw	Item 6	455-861	---	C	---	---	No
Jack Nut	Item 7	455-861	---	C	---	---	No
Jam Nut	Item 8	455-861	---	C	---	---	No
Base Weldment Inlet Cover	Item 10	455-861	SR, TH, SH	B	X	---	Yes
Base Weldment Shield Ring	Item 11	455-861	SR, TH, SH	B	X	---	Yes
Base Weldment Bottom Plate	Item 12	455-861	SR, TH, SH	B	X	---	Yes
Inlet Side	Item 13	455-861	SR, TH, SH	B	X	---	Yes
Inlet Top	Item 14	455-861	SR, TH, SH	B	X	---	Yes
Stand Plate	Item 15	455-861	SR, TH, SH	B	X	---	Yes
Baffle Weldment Base Plate	Item 16	455-861	SR, TH, SH	B	X	---	Yes
Nelson Stud	Item 17	455-861	SR	B	X	---	Yes
Outlet Bottom	Item 18	455-861	SR, TH, SH	B	X	---	Yes
Outlet Top	Item 19	455-861	SR, TH, SH	B	X	---	Yes
Outlet Shield Plate	Item 20	455-861	SR, TH, SH	B	X	---	Yes
Outlet Bottom	Item 21	455-861	SR, TH, SH	B	X	---	Yes
Outlet Top	Item 22	455-861	SR, TH, SH	B	X	---	Yes
Outlet Side	Item 23	455-861	SR, TH, SH	B	X	---	Yes
Outlet Back	Item 24	455-861	SR, TH, SH	B	X	---	Yes
Baffle	Item 25	455-861	SR, TH, SH	B	X	---	Yes
Square Nut	Item 26	455-861	---	NQ	---	---	No
Heavy Hex Nut	Item 27	455-861	---	NQ	---	---	No

Table 2.5-4 Intended Functions of NAC-MPC Vertical Concrete Cask (VCC) Subcomponents YR-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Primer and Coating for Liner, Pedestal and Baseplate Assemblies	Note 3	455-861	---	NQ	---	---	No
Lid Bolt	Item 6	455-862	SR	B	X	---	Yes
Washer	Item 7	455-862	---	NQ	---	---	No
Insulation	Item 8	455-862	---	NQ	---	---	No
Cover	Item 9	455-862	SR	B	X	---	Yes
Seal Tape	Item 10	455-862	---	NQ	---	---	No
Seal Wire	Item 11	455-862	---	C	---	---	No
Security Seal	Item 12	455-862	---	C	---	---	No
Tab	Item 13	455-862	---	NQ	---	---	No
VCC Lid	Item 1	455-863	SR	B	X	---	Yes
Coating System for VCC Lid	Note 1	455-863	---	NQ	---	---	No
Shield Plug Plate	Item 1	455-864	SR	B	X	---	Yes
Neutron Shield Retaining Ring	Item 2	455-864	SR	B	X	---	Yes
Neutron Shielding	Item 3	455-864	SH	B	X	---	Yes
Neutron Shield Cover Plate	Item 4	455-864	SR	B	X	---	Yes
Coating System for VCC Shield Plug	Item 5 and Dwg. Note 1	455-864	---	NQ	---	---	No
Rebar	Items 1-11	455-866	SR, SH	B	X	---	Yes
Concrete Shell	Item 15	455-866	SR, SH	B	X	---	Yes
Screen Strips	Item 16	455-866	---	C	---	---	No
Vent Screen	Item 17, 21, 22	455-866	---	C	---	---	No
Screen Bolt	Item 19	455-866	---	NQ	---	---	No
Plain Washer	Item 20	455-866	---	NQ	---	---	No
Concrete Anchor	Item 23	455-866	---	NQ	---	---	No
Lag Bolt	Item 24	455-866	---	NQ	---	---	No

Table 2.5-4 Intended Functions of NAC-MPC Vertical Concrete Cask (VCC) Subcomponents YR-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Sealer	Item 25	455-866	---	NQ	---	---	No
VCC Inlet Supplemental Shield Side Plate	Item 1	455-913	SH	B	X	---	Yes
Shield Pipe	Item 2	455-913	SH	B	X	---	Yes
Coating System for VCC Supplemental Shield	Item 3 and Dwg. Note 1	455-913	---	NQ	---	---	No
Shims	Item 4	455-913	---	NQ	---	---	No
VCC Nameplate	Item 1	455-856	---	NQ	---	---	No

Notes:

(1) Included in Section 1.7 of the NAC-MPC System Updated Final Safety Analysis Report (FSAR) [2.7.1.a – 2.7.1.m]
(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Subcriticality (CR), and Retrievability (RE)
(3) Items identified as No in the In-Scope column do not have an identified ITS function and do not require aging management review.
(4) Non-Quality (NQ) is used for NITS designation.

Table 2.5-5 Intended Functions of NAC-MPC Vertical Concrete Cask (VCC) Subcomponents CY-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
VCC Liner Shell	Item 1	414-861	SH, TH, SR	B	X	---	Yes
Top Flange	Item 2	414-861	SR	B	X	---	Yes
Support Ring	Item 3	414-861	SR	B	X	---	Yes
Jack Base	Item 4	414-861	---	NQ	---	---	No
Jack Gusset	Item 5	414-861	---	NQ	---	---	No
Jack Screw	Item 6	414-861	---	NQ	---	---	No
Jack Nut	Item 7	414-861	---	NQ	---	---	No
Jam Nut	Item 8	414-861	---	NQ	---	---	No
Base Weldment Inlet Cover	Item 10	414-861	SR, TH, SH	B	X	---	Yes
Base Weldment Shield Ring	Item 11	414-861	SR, TH, SH	B	X	---	Yes
Base Weldment Bottom Plate	Item 12	414-861	SR, TH, SH	B	X	---	Yes
Inlet Side	Item 13	414-861	SR, TH, SH	B	X	---	Yes
Inlet Top	Item 14	414-861	SR, TH, SH	B	X	---	Yes
Stand Plate	Item 15	414-861	SR, TH, SH	B	X	---	Yes
Baffle Weldment Base Plate	Item 16	414-861	SR, TH, SH	B	X	---	Yes
Nelson Stud	Item 17	414-861	SR	B	X	---	Yes
Outlet Bottom	Item 18	414-861	SR, TH, SH	B	X	---	Yes
Outlet Top	Item 19	414-861	SR, TH, SH	B	X	---	Yes
Outlet Shield Plate	Item 20	414-861	SH, TH	B	X	---	Yes
Outlet Bottom	Item 21	414-861	SR, TH, SH	B	X	---	Yes
Outlet Top	Item 22	414-861	SR, TH, SH	B	X	---	Yes
Outlet Side	Item 23	414-861	SR, TH, SH	B	X	---	Yes
Outlet Back	Item 24	414-861	SR, TH, SH	B	X	---	Yes
Baffle Weldment	Item 25	414-861	SR, TH, SH	B	X	---	Yes
Square Nut	Item 26	414-861	---	NQ	---	---	No
Cover	Item 27	414-861	SR	B	X	---	Yes

Table 2.5-5 Intended Functions of NAC-MPC Vertical Concrete Cask (VCC) Subcomponents CY-MPC

Subcomponent	Part or I.D. No.	Reference Drawing [(1)]	Intended Safety Functions [(2)]	Quality Category	Sub-Scoping Results		In-Scope [(3)]
					Criterion 1	Criterion 2	
Dowel Pins	Item 28	414-861	---	NQ	---	---	No
Lifting Nut	Item 29	414-861	---	NQ	---	---	No
Primer and Paint for Liner, Pedestal and Baseplate Assemblies	Items 30 and 31, and Dwg. Note 3	414-861	---	NQ	---	---	No
Security Seal	Item 3	414-862	---	NQ	---	---	No
Lid Bolt	Item 6	414-862	SR	B	X		Yes
Washer	Item 7	414-862	---	NQ	---	---	No
Seal Tape	Item 10	414-862	---	NQ	---	---	No
Seal Wire	Item 11	414-862	---	C	---	---	No
VCC Lid	Item 1	414-863	SR	B	X		Yes
Primer and Paint for VCC Lid	Items 2 and 3, and Dwg. Note 1	414-863	---	NQ	---	---	No
Shield Plug Plate	Item 1	414-864	SR, SH	B	X	---	Yes
Neutron Shield Retaining Ring	Item 2	414-864	SR	B	X	---	Yes
Neutron Shield Cover Plate	Item 3	414-864	SR, SH	B	X	---	Yes
Lifting and Center Boss	Item 4 & 7	414-864	SR	NQ	---	---	No
Neutron Shielding	Items 5 & 6	414-864	SH	B	X	---	Yes
Primer and Paint for Shield Plug	Items 8 and 9, and Dwg. Note 1	414-864	---	NQ	---	---	No
VCC Rebar	Items 1-11	414-866	SR, SH	B	X	---	Yes
Concrete Shell	Item 15	414-866	SR, SH	B	X	---	Yes
Vent Screen	Items 16 & 30	414-866	---	C	---	---	No
Vent Strips	Item 17	414-866	---	C	---	---	No

Table 2.5-5 Intended Functions of NAC-MPC Vertical Concrete Cask (VCC) Subcomponents CY-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Screen Bolt	Item 19	414-866	---	NQ	---	---	No
Concrete Anchor	Item 22, 26 & 31	414-866	---	NQ	---	---	No
Flat Washer	Item 23	414-866	---	NQ	---	---	No
Lag Bolt	Item 24	414-866	---	NQ	---	---	No
Sealer	Item 25	414-866	---	NQ	---	---	No
Screen Bolt	Item 27	414-866	---	NQ	---	---	No
Washer	Item 28	414-866	---	NQ	---	---	No
Retainer Plate	Item 29	414-866	---	NQ	---	---	No
Nameplate	Item 1	414-856	---	NQ	---	---	No
Black Weather Resistant Paint	Item 2	414-856	---	NQ	---	---	No

Notes:

(1) Included in Section 1.7 of the NAC-MPC System Updated Final Safety Analysis Report (FSAR) [2.7.1.a – 2.7.1.m]
(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Subcriticality (CR), and Retrievability (RE)
(3) Items identified as No in the In-Scope column do not have an identified ITS function and do not require aging management review.
(4) Non-Quality (NQ) is used for NITS designation.

Table 2.5-6 Intended Functions of NAC-MPC Vertical Concrete Cask (VCC) Subcomponents MPC-LACBWR

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
VCC Liner Shell	Item 1	630045-861	SH, TH, SR	B	X	---	Yes
Top Flange	Item 2	630045-861	SR	B	X	---	Yes
Weldment Bottom Plate	Item 4	630045-861	SR, TH, SH	B	X	---	Yes
Inlet Side Plate	Item 5	630045-861	SR, TH, SH	B	X	---	Yes
Inlet Top Plate	Item 6	630045-861	SR, TH, SH	B	X	---	Yes
Stand Base Plate	Item 7	630045-861	SR, TH, SH	B	X	---	Yes
Base Plate	Item 8	630045-861	SR, TH, SH	B	X	---	Yes
Nelson Stud	Item 9	630045-861	SR	B	X	---	Yes
Outlet Bottom Plate	Item 10	630045-861	SR, TH, SH	B	X	---	Yes
Outlet Top Plate	Item 11	630045-861	SR, TH, SH	B	X	---	Yes
Outlet Shield Plate	Item 12	630045-861	SH	B	X	---	Yes
Outlet Bottom	Item 13	630045-861	SR, TH, SH	B	X	---	Yes
Outlet Top	Item 14	630045-861	SR, TH, SH	B	X	---	Yes
Outlet Side	Item 15	630045-861	SR, TH, SH	B	X	---	Yes
Outlet Back	Item 16	630045-861	SR, TH, SH	B	X	---	Yes
Baffle Weldment	Item 17	630045-861	SR, TH, SH	B	X	---	Yes
Screen Tab	Item 18	630045-861	---	NQ	---	---	No
Dowel Pin	Item 19	630045-861	---	NQ	---	---	No
Primer and Paint for Liner, Pedestal and Baseplate Assemblies	Item 20 and Dwg. Note 3	630045-861	---	NQ	---	---	No
Inlet Shield Pipe/Tube/Bar	Item 21	630045-861	SH	B	X	---	Yes
Baffle Coverplate	Item 22	630045-861	SR	C	X	---	Yes
Lid Bolt	Item 3	630045-862	SR	B	X	---	Yes
Washer	Item 4	630045-862	---	NQ	---	---	No

Table 2.5-6 Intended Functions of NAC-MPC Vertical Concrete Cask (VCC) Subcomponents MPC-LACBWR

Subcomponent	Part or I.D. No.	Reference Drawing [(1)]	Intended Safety Functions [(2)]	Quality Category	Sub-Scoping Results		In-Scope [(3)]
					Criterion 1	Criterion 2	
VCC Lid Bottom Plate	Item 1	630045-863	SR, SH	B	X	---	Yes
Lid Ring	Item 2	630045-863	SR	B	X	---	Yes
VCC Lid Top Plate	Item 3	630045-863	SR, SH	B	X	---	Yes
Concrete	Item 4	630045-863	SH	B	X	---	Yes
Center Support	Item 5	630045-863	SR	B	X	---	Yes
Nelson Stud	Item 6	630045-863	SR	B	X	---	Yes
Primer and Paint for VCC Lid	Item 7 and Dwg. Note 1	630045-863	---	NQ	---	---	No
VCC Nameplate	Item 1	630045-864	---	NQ	---	---	No
Black Paint	Item 2 and Dwg. Note 4	630045-864	---	NQ	---	---	No
VCC Rebar	Items 1, 2, 4-11, 26 & 27	630045-866	SR, SH	B	X	---	Yes
RTD Mounting Plate	Item 3	630045-866	---	NQ	---	---	No
Concrete Shell	Item 15	630045-866	SR, SH	B	X	---	Yes
Screen Strips	Item 16	630045-866	---	NQ	---	---	No
Vent Screen	Item 17	630045-866	---	NQ	---	---	No
Screen Bolt	Item 19	630045-866	---	NQ	---	---	No
Plain Washer	Item 20	630045-866	---	NQ	---	---	No
Concrete Anchors	Item 23	630045-866	---	NQ	---	---	No
Cap Screw	Item 24	630045-866	---	NQ	---	---	No
Sealer	Item 25	630045-866	---	NQ	---	---	No
Retainer Plate	Item 28	630045-866	---	NQ	---	---	No
Inlet Screen	Item 29	630045-866	---	NQ	---	---	No
Screen Bolt	Item 30	630045-866	---	NQ	---	---	No

Table 2.5-6 Intended Functions of NAC-MPC Vertical Concrete Cask (VCC) Subcomponents MPC-LACBWR

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Resistance Temperature Detector (RTD)	Item 31	630045-866	---	NQ	---	---	No
RTD Connection Head	Item 32	630045-866	---	NQ	---	---	No

Notes:

1. (1) Included in Section 1.A.7 of the NAC-MPC System Updated Final Safety Analysis Report (FSAR) [2.7.1.a – 2.7.1.m]
2. (2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Subcriticality (CR), and Retrievability (RE)
3. (3) Items identified as No in the In-Scope column do not have an identified ITS function and do not require aging management review.
4. (4) Non-Quality (NQ) is used for NITS designation.

Table 2.5-7 Intended Functions of NAC-MPC Transfer Cask (TFR) Subcomponents YR-MPC and MPC-LACBWR

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Bottom Plate	Item 1	455-860	SR	B	X	---	Yes
Inner Shell	Item 2	455-860	SR	B	X	---	Yes
Gamma Shield Brick	Items 3 and 22	455-860	SH	B	X	---	Yes
Outer Shell	Item 4	455-860	SR	B	X	---	Yes
Trunnion	Item 5	455-860	SR	B	X	---	Yes
Trunnion Cap	Item 6	455-860	---	B	---	---	No
Scuff Plate	Item 7	455-860	---	C	---	---	No
Neutron Shield	Item 8	455-860	SH	B	X	---	Yes
Top Plate	Item 9	455-860	SR	B	X	---	Yes
Door Rail	Item 10	455-860	SR, SH	B	X	---	Yes
Shield Door A (3)	Item 11	455-860	SR, SH	B	X	---	Yes
Shield Door B (3)	Item 12	455-860	SR, SH	B	X	---	Yes
Door Lock Bolt	Item 13	455-860	SR	C	X	---	Yes
Retaining Ring (3)	Item 14	455-860	SR	B	X	---	Yes
Retaining Ring Bolt (3)	Item 15	455-860	SR	B	X	---	Yes
Connector	Item 17	455-860	SR	C	X	--	Yes
Fill/Drain Line Plate	Item 18	455-860	---	C	---	---	No
Fill/Drain Line Pipe	Item 16	455-860	---	C	---	---	No
Spent Fuel Pool Compatible Coating System	Item 23 and Dwg. Note 7	455-860	---	NQ	---	---	No
Lubricant	Item 24 and Dwg. Note 8	455-860	---	NQ	---	---	No
Lead Wool	Item 25	455-860	---	NQ	---	---	No

Table 2.5-7 Intended Functions of NAC-MPC Transfer Cask (TFR) Subcomponents YR-MPC and MPC-LACBWR

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Black Weather Resistant Paint (for Component ID)	Item 26 and Dwg. Note 15	455-860	---	NQ	---	---	No
Nameplate	Item 27	455-860	---	NQ	---	---	No
Dowel Pin	Item 28	455-860	---	NQ	---	---	No
Door Lock Bolt	Item 29	455-860	SR	C	X	---	Yes
Flat Washer (5)	Item 31	455-860	---	NQ	---	---	No
Safety Wire (5)	Item 32	455-860	---	NQ	---	---	No
Strut Bracket (5)	Item 33	455-860	SR	B	X	---	Yes
Hex Head Bolt (5)	Item 34	455-860	SR	B	X	---	Yes
Lock Pin	Item 5	455-918	SR	NQ	---	X	Yes
Door Stop	Items 1-4 & 6	455-918	---	NQ	---	---	No
Transfer Adapter	Items 1 - 5	455-859	SH	C	X	---	Yes

Notes:

(1) Included in Section 1.7 of the NAC-MPC System Updated Final Safety Analysis Report (FSAR) [2.7.1.a – 2.7.1.m]
(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Subcriticality (CR), and Retrievability (RE)
(3) Items identified as No in the In-Scope column do not have an identified ITS function and do not require aging management review.
(4) Identified original components were removed and disposed of. Replacement components were provided in accordance with NAC Drawing No. 630045-060 as listed above.
(5) Identified items designed for TSC transfer at YR and removed for operations at LACBWR. Items are no longer available.
(6) Non-Quality (NQ) is used for NITS designation.

Table 2.5-8 Intended Functions of NAC-MPC Transfer Cask (TFR) Subcomponents CY-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Bottom Plate	Item 1	414-860	SR	B	X	---	Yes
Inner Shell	Item 2	414-860	SR	B	X	---	Yes
Gamma Shield Brick	Item 3	414-860	SH	B	X	---	Yes
Outer Shell	Item 4	414-860	SR	B	X	---	Yes
Trunnion	Item 5	414-860	SR	B	X	---	Yes
Trunnion Cap	Item 6	414-860	---	NQ	---	---	No
Scuff Plate	Item 7	414-860	---	NQ	---	---	No
Neutron Shield	Item 8	414-860	SH	B	X	---	Yes
Top Plate	Item 9	414-860	SR	B	X	---	Yes
Door Rail	Item 10	414-860	SR, SH	B	X	---	Yes
Shield Door A	Item 11	414-860	SR, SH	B	X	---	Yes
Shield Door B	Item 12	414-860	SR, SH	B	X	---	Yes
Door Lock Bolt	Item 13	414-860	SR	C	X	---	Yes
Retaining Ring	Item 14	414-860	SR	B	X	---	Yes
Retaining Ring Bolt	Item 15	414-860	SR	B	X	---	Yes
Connector	Item 17	414-860	SR	C	X	---	Yes
Fill/Drain Line Plate	Item 20	414-860	---	C	---	---	No
Fill/Drain Line Pipe	Item 21	414-860	---	C	---	---	No
Spent Fuel Pool Compatible Coating System	Item 22 and Dwg. Note 7	414-860	---	NQ	---	---	No
Spent Fuel Pool Compatible Lubricant	Item 23	414-860	---	NQ	---	---	No
Black Weather Resistant Paint (for Component ID)	Item 24 and Dwg. Note 13	414-860	---	NQ	---	---	No

Table 2.5-8 Intended Functions of NAC-MPC Transfer Cask (TFR) Subcomponents CY-MPC

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Commercial Grade Lead Wool	Item 25	414-860	---	NQ	---	---	No
Nameplate	Item 26	414-860	---	NQ	---	---	No
Dowel Pin	Item 27	414-860	---	NQ	---	---	No
Lock Pin	Item 5	414-917	SR	NQ	---	X	Yes
Door Stop	Item 1-4 & 6	414-917	---	NQ	---	---	No
Transfer Adapter	Items 1 - 5	455-859	SH	C	X	---	Yes

Notes:

(1) Included in Section 1.7 of the NAC-MPC System Updated Final Safety Analysis Report (FSAR) [2.7.1.a – 2.7.1.m]

(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Subcriticality (CR), and Retrievability (RE)

(3) Items identified as No in the In-Scope column do not have an identified ITS function and do not require aging management review.

(4) Non-Quality (NQ) is used for NITS designation.

Table 2.5-9 Intended Functions of Spent Fuel Assembly[(1)] (SFA) Subcomponents in NAC-MPC Systems

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Category	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Fuel rod cladding	NA	NA	CO, CR, RE, SH, SR, TH	A	X	---	Yes
Guide tubes (PWR) or water channels (BWR)	NA	NA	RE, SR	A	X	---	Yes
Spacer grids	NA	NA	CR, RE, SR, TH	A	X	---	Yes
Lower and upper end fittings	NA	NA	CR, RE, SR	A	X	---	Yes
Fuel channel (BWR)	NA	NA	CR, TH	A	X	---	Yes
Poison rod assemblies (PWR)	NA	NA	CR	A	X	---	Yes

Notes:

(1) SFA for NAC-MPC Systems described in Sections 1.3.1 and 1.A.3 of the NAC-MPC FSAR [2.7.1.a – 2.7.1.m]

(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Subcriticality (CR), and Retrievability (RE)

(3) The NAC-MPC criticality analysis does not account for negative reactivity effects of control components. Therefore, the control components do not have a criticality control function.

2.7 REFERENCES

2.7.1 NAC International, Inc., "Final Safety Analysis Report for the NAC-MPC Multi-Purpose Canister System," Docket No. 72-1025,

- 2.7.1.a NAC-MPC System FSAR, Revision 0, May 2000
- 2.7.1.b NAC-MPC System FSAR, Revision 1, February 2002
- 2.7.1.c NAC- MPC System FSAR, Revision 2, November 2002
- 2.7.1.d NAC- MPC System FSAR, Revision 3, March 2004
- 2.7.1.e NAC- MPC System FSAR, Revision 4, November 2004
- 2.7.1.f NAC- MPC System FSAR, Revision 5, October 2005
- 2.7.1.g NAC- MPC System FSAR, Revision 6, November 2006
- 2.7.1.h NAC- MPC System FSAR, Revision 7, November 2008
- 2.7.1.i NAC- MPC System FSAR, Revision 8, February 2009
- 2.7.1.j NAC- MPC System FSAR, Revision 9, November 2010
- 2.7.1.k NAC- MPC System FSAR, Revision 10, January 2014
- 2.7.1.l NAC- MPC System FSAR, Revision 11, April 2018
- 2.7.1.m NAC- MPC System FSAR, Revision 12, April 2019

2.7.2 U.S. Nuclear Regulatory Commission, Certificate of Compliance for Spent Fuel Storage Casks, Model No.:

- 2.7.2.a NAC-MPC CoC; Initial Issue Revision 0, Effective April 10, 2000.
- 2.7.2.b NAC-MPC CoC; Amendment No. 1, Effective November 13, 2001.
- 2.7.2.c NAC-MPC CoC; Amendment No. 2, Effective May 29, 2002.
- 2.7.2.d NAC-MPC CoC; Amendment No. 3, Effective October 1, 2003.
- 2.7.2.e NAC-MPC CoC; Amendment No. 4, Effective October 27, 2004.
- 2.7.2.f NAC-MPC CoC; Amendment No. 5, Effective July 24, 2007.
- 2.7.2.g NAC-MPC CoC; Amendment No. 6, Effective October 4, 2010.
- 2.7.2.h NAC-MPC CoC; Amendment No. 7, Effective March 4, 2019.
- 2.7.2.i NAC-MPC CoC; Amendment No. 8, Effective March 4, 2019.

2.7.3 Safety Evaluation Report (SER) for NAC-MPC System Certificate of Compliance No. 1025,

- 2.7.3.a SER for NAC-MPC System CoC, Revision 0, March 10, 2000
- 2.7.3.b SER for NAC-MPC System CoC, Revision 1, January 23, 2002
- 2.7.3.c SER for NAC-MPC System CoC, Revision 2, May 30, 2002
- 2.7.3.d SER for NAC-MPC System CoC, Revision 3, October 8, 2003
- 2.7.3.e SER for NAC-MPC System CoC, Revision 4, October 27, 2004
- 2.7.3.f SER for NAC-MPC System CoC, Revision 5, July 24, 2007
- 2.7.3.g SER for NAC-MPC System CoC, Revision 6, October 4, 2010 .
- 2.7.3.h SER for NAC-MPC System CoC, Revision 7, March 4, 2019
- 2.7.3.i SER for NAC-MPC System CoC, Revision 8, March 4, 2019

2.7.4 NEI 14-03, Revision 2, “Guidance for Operations-Based Aging Management for Dry Cask Storage”, December 2016

2.7.5 NAC-STC Safety Analysis Report (SAR), Revision 20, July 31, 2019.

2.7.6 NRC Certificate of Compliance for NAC-STC Transport Cask, Docket 71-9235, CoC No. 9253, Revision 22, July 8, 2019.

2.7.7 NUREG-1927, Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance, Revision 1, June 2016

2.7.8 Federal Register, Volume 60, No. 88, Page 22464, dated May 8, 1995, Nuclear Power Plant License Renewal; Revisions, 10 CFR Parts 2, 51, and 54

2.7.9 NUREG-0612, Control of Heavy Loads at Nuclear Power Plants

2.7.10 ANSI N14.6, American National Standard for Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500kg) or More for Nuclear Materials

3.0 AGING MANAGEMENT REVIEWS

The Aging Management Review (AMR) of the NAC-MPC System provides an assessment of the aging effects that could adversely affect the ability of the in-scope SSCs to perform their intended function during the period of extended operation. The scoping process identified the NAC-MPC System SSCs within the scope of CoC renewal which require evaluation for the effects of aging in the aging management review process. The methodology used for the AMR of the NAC-MPC System is based on the guidance provided in NUREG-1927 [3.9.2].

The purpose of the AMR process is to assess the in-scope NAC-MPC System SSCs with respect to aging effects that could affect the ability of the SSC to perform its intended function during the period of extended operation. The aging management review process involves the following five (5) major steps:

1. Identification of the materials and environments for all subcomponents of the in-scope SSC.
2. Identification of aging effects requiring management during the period of extended operation.
3. Identification and evaluation of the time limited aging analyses (TLAAs) for the extended storage period.
4. Identification of aging management programs (AMPs) for managing aging effects during the period of extended operation.
5. Evaluation of fuel retrievability during the period of extended operation.

Identification of the subcomponents of in-scope SSC requiring AMR and the identification of the materials and environments for all in-scope SSC are discussed in Sections 3.1. Aging effects that require management during the period of extended operation are discussed in Section 3.2. In-scope SSC that are determined to be subject to an aging effect that could adversely affect their ability to perform their safety function(s) are required to either be evaluated with Time-Limited Aging Analysis (TLAA) or to be managed through an existing, modified, or new Aging Management Program (AMP). The TLAA evaluations and AMP used to manage aging effects on the in-scope SSC are discussed in Sections 3.3 and 3.4, respectively. Periodic tollgate assessment reviews are discussed in Section 3.5, and fuel retrievability during the period of extended operation is evaluated in Section 3.6. A summary of the NAC-MPC System operating experience is presented in Section 3.7 and a discussion of the design basis document review efforts are presented in Section 3.8. The results of the AMR are summarized in Tables 3.2-1 through 3.2-9. References for this section are provided in Section 3.9.

3.1 IDENTIFICATION OF SSC MATERIALS AND ENVIRONMENTS

The scoping process completed in Section 2 identified the specific SSC subcomponents for the in-scope NAC-MPC System SSCs that require aging management review (AMR), although they do not identify potential aging effects or mechanisms, or specific aging management methods. The in-scope SSCs and their intended safety functions are identified in Tables 2.5-1 thru 2.5-9. Therefore, the first step of the AMR process is to further review the in-scope SSCs to identify and describe the SSC subcomponents that support the intended function of the in-scope SSCs.

The materials of construction for the in-scope SSC and their associated subcomponents are identified by reviewing the NAC License Drawings contained in the NAC-MPC System FSARs [3.9.1.a thru 3.9.1.m] and the documentation listed in Section 3.8. The environments to which the materials are normally exposed are identified based on a review of the latest NAC-MPC System FSAR [3.9.1.m], and plant loading procedures and records, and are defined and classified in accordance with the environments defined in NUREG-2214, “Managing Aging Processes in Storage (MAPS) Report” [3.9.4]. The materials of construction and environments for each of the in-scope SSC are discussed in Section 3.1.1 and 3.1.2, respectively, and summarized in Tables 3.2-1 through 3.2-10. The combinations of materials and environments are used to identify the potential aging effects that require management during the period of extended operation and are discussed in Section 3.2.

3.1.1 Identification of In-Scope SSC Subcomponent Materials

The second step of the aging management review process is the identification of the materials of construction the SSC subcomponents that require an aging management review. The materials of construction were identified through a review of pertinent design and/or design basis documents, which are discussed in Subsection 3.8.

3.1.1.1 Transportable Storage Canister (TSC) and Fuel Baskets

The TSC is the main component of the NAC-MPC System and is available in three different lengths to accommodate various lengths of PWR and BWR fuel assemblies and non-fuel components. The TSC provides for the safe storage and leak tight confinement of the radioactive materials contained in the stored spent fuel and prevents their release to the environment under all normal and accident conditions of storage. The TSC assembly consists of an all-welded stainless-steel canister that contains a PWR or BWR fuel basket structure and the spent fuel assembly contents. The TSC vessel has been designed, fabricated and inspected in accordance with the ASME Code, Section III, Subsection NB, to the maximum practical extent, with NRC approved exemptions.

The major components of the YR-MPC and CY-MPC TSC vessel are the shell, base plate, shield lid, port covers and structural lid. The field installed and welded shield lid, vent and drain port covers, and structural lid provide the redundant (primary and secondary) confinement closure system. The shield lid also provides radiological shielding for operations personnel performing the cask preparation activities (e.g., TSC cavity draining, vacuum drying, lid welding, and pressure and leakage testing). Threaded holes in the TSC structural lid are provided for attachment of lifting hoist rings and slings to lift and handle the loaded TSC.

The major components of the MPC-LACBWR TSC vessel are the shell, base plate, closure lid, inner and outer port cover plates, and closure ring similar to the NAC MAGNASTOR TSC design. The field installed and welded closure lid, inner and outer vent and drain port covers, and closure ring provide the redundant (primary and secondary) confinement closure system. The closure lid also provides radiological shielding for operations personnel performing the cask preparation activities (e.g., TSC cavity draining, vacuum drying, lid welding, and pressure and leakage testing). A 4.0-inch-thick 6061-T651 aluminum spacer plate is bolted to the underside of the closure lid to limit the space between the lid and the undamaged fuel assemblies (e.g., without DFCs). Threaded holes in the TSC closure lid are provided for attachment of lifting hoist rings and slings to lift and handle the loaded TSC.

The TSC shell is fabricated from a cylindrically rolled, ⅝-inch-thick (0.625 in.) [YR and CY] and ½-inch thick (0.50 in.) [MPC-LACBWR] SA240, Type 304L stainless steel plate. The nominal external diameter of the TSC shell is 70.64 inches with a 69.39-inch nominal internal diameter. The shell is formed with a full penetration weld. If the TSC shell required a girth weld, the seam welds of adjacent shell sections were offset approximately 45º. The TSC shell seam and girth welds were nondestructively examined (NDE) using radiographic examination (RT) methods in accordance with the ASME Code, Section V, Article 2, with weld acceptance criteria per Section III, Subsection NB, Article NB-5320.

Following acceptance of the shell weldment, it was welded to a SA240, Type 304L stainless steel, base plate (1.0 in. thick for YR-MPC, 1.75 in. for CY-MPC, and 1.25 in. for MPC-LACBWR) with a full penetration weld. The NDE of the TSC shell to the base plate weld was performed using the ultrasonic examination (UT) method in accordance with the ASME Code, Section V, Article 5, with weld acceptance criteria per Section III, Subsection NB, Article NB-5330. Located and welded to the inside surface of the base plate are four ASTM A240/A276, Type 304 stainless steel location lugs. These location lugs are provided to locate, align and prevent rotation of the basket structure assembly during use. The lugs interface with the bottom weldment of the basket assembly. The TSC shell assembly is cleaned, and the appropriate PWR or BWR basket assembly was installed and aligned using the location lugs. To secure the basket assembly axially in the TSC shell assembly, and to position the TSC shield lid for welding (i.e., closure lid for MPC-LACBWR), a SA479/SA240, Type 304 stainless steel, ½ x ½-inch-square lid support ring was installed, positioned and welded to the TSC shell above the basket assembly top weldment. Additionally, an ASTM A240/A276, Type 304 stainless steel 4-½-inch-long x 1-inch-wide x ½-inch-high key was welded in the 1-inch gap in the lid support ring. The key and support ring are provided to align and vertically position the TSC shield lid (i.e., closure lid for MPC-LACBWR).

For each YR-MPC and CY-MPC TSC shell assembly, a unique TSC shield lid, structural lid, port covers, and drain tube assembly was fabricated. The TSC shield lid is a SA240/SA182, Type 304 stainless steel, 5-inch-thick, 69.0-inch-diameter plate/forging that is installed on a loaded TSC assembly underwater in the spent fuel pool. The shield lid rests on the lid support ring and is rotationally aligned by the key. Following removal of the TFR from the pool, the TSC is prepared and the TSC shield lid was welded to the TSC shell with a ½-inch-thick, multi-pass partial penetration weld. NDE of the TSC shield lid-to-TSC shell weld was performed using root and final surface visual (VT) and dye penetrant (PT) examination methods in accordance with the

ASME Code, Section V, Article 6, with weld acceptance criteria per Section III, Subsection NB, Article NB-5350. As required, SA240/A240, Type 304 stainless steel shims were used to reduce the weld gap during shield lid-to-TSC shell welding operation.

The YR-MPC and CY-MPC TSC shield lids are each provided with two 1-inch-diameter fitting penetrations through the lid for the vent and drain openings. The vent opening is provided with a self-sealing, quick-disconnect valved nipple. At the drain opening, an identical valved nipple is attached to a Type 304 stainless steel 1-inch-diameter tube, which is inserted through the TSC shield lid and basket assembly to approximately ⅛-inch from the bottom of the canister. The drain and vent valved nipples are sealed to the TSC shield lid threaded openings using stainless steel metal, Viton or EDPM polymer seals. The quick-disconnect valved nipples are operated using connector assemblies with mating female self-sealing valves. The vent and drain openings are utilized during loaded TSC preparation activities to provide access to the TSC cavity for water draining/blowdown operations, vacuum drying, pneumatic pressure testing, helium backfilling and helium leakage testing. The vent and drain openings are also designed for use during TSC unloading operations to provide access to the cavity for water filling/cooldown operations of the TSC and its contents. No confinement credit is taken by the quick-disconnect valved nipples during storage operations.

Following pressure testing, draining, drying and backfilling of the cavity with helium, the vent and drain openings were closed by welding in place SA479, Type 304 stainless steel, ½-inch thick x 5.9-inch diameter port covers that fit around the valved nipple and fill the penetration volume to minimize streaming. The port covers were welded to the shield lid using a partial penetration weld. NDE of the port cover-to-shield lid welds is performed by PT examination of the final pass in accordance with the ASME Code, Section V, Article 6, with weld acceptance criteria per Section III, Subsection NB, Article NB-5350. At the completion of the confinement boundary, as defined by the shield lid-to-shell, and port cover-to-lid welds, the boundary was tested for helium leakage to leak-tight criteria in accordance with ANSI N14.5 [3.9.26] requirements. The TSC shield lid is provided with three, 1-8UNC-2B threaded holes for installation of lifting hoist rings for handling of the shield lid. Optional stainless steel threaded plugs may be installed flush in the shield lid threaded holes to minimize radiation streaming effects during storage.

Following closure, welding and testing of the TSC shield lid, the YR-MPC or CY-MPC TSC structural lid was installed on top of the shield lid. The TSC structural lid is a SA240/SA182, Type 304L 3-inch-thick, 68.7-inch- diameter stainless steel plate/forging. A SA479/SA240, Type 304 ½ x ½-inch stainless steel spacer ring was installed in a machined groove around the structural lid. The spacer ring provides proper fit-up and fills the gap between the structural lid and the TSC shell. The TSC structural lid-to-TSC shell weld is a ⅞-inch multi-pass partial penetration weld performed with progressive VT and PT examinations of the root, each intermediate weld layer (not exceeding ⅜-inch), and the final weld surface. The PT examinations were performed in accordance with the ASME Code, Section V, Article 6, with weld acceptance criteria per Section III, Subsection NB, Article NB-5350. The TSC structural lid is provided with six 2–4 ½ UNC-2B threaded holes for engagement of lifting hoist rings or other handling components and are designed for the single-failure-proof handling of the loaded and closed TSC.

For each MPC-LACBWR TSC shell assembly, a unique TSC closure lid, inner and outer port cover plates, closure ring, and drain tube assembly were fabricated. The MPC-LACBWR TSC closure lid is a SA240/SA182, Type 304/304L, 7-inch-thick, 69.39-inch- diameter stainless steel plate/forging. The TSC closure lid was installed on a loaded TSC assembly underwater in the spent fuel pool, rests on the lid support ring and was rotationally aligned by the key. Following removal of the TFR from the pool, the TSC was prepared and the TSC closure lid welded to the TSC shell with a ½-inch-thick, multi-pass partial penetration weld. NDE of the TSC closure lid-to-TSC shell weld was performed using root, mid-plane and final surface visual (VT) and dye penetrant (PT) examination methods in accordance with the ASME Code, Section V, Article 6, with weld acceptance criteria per Section III, Subsection NB, Article NB-5350. As required, SA240/A240, Type 304 stainless steel shims were used to reduce the weld gap during closure lid-to-TSC shell welding operation.

The MPC-LACBWR TSC closure lid is provided with two 1-inch-diameter fitting penetrations for the vent and drain openings. The vent opening is provided with a self-sealing, quick-disconnect valved nipple. At the drain opening, an identical valved nipple is attached to a Type 304 stainless steel 1-inch-diameter tube, which is inserted through the TSC closure lid and basket assembly to approximately ⅛-inch from the bottom of a 3-inch diameter x ⅜-inch recess in the base plate. The inclusion of the recess in the TSC base plate will allow more of the cavity water inventory to be removed by pumping or blowdown operations. The drain and vent valved nipples are sealed to the TSC shield lid threaded openings using Viton seals. The quick-disconnect valved nipples are operated using connector assemblies with mating female self-sealing valves. The vent and drain openings are utilized during loaded TSC preparation activities to provide access to the TSC cavity for water draining/blowdown operations, vacuum drying, pressure testing, helium backfilling and helium leakage testing. The vent and drain openings are also designed for use during TSC unloading operations to provide access to the cavity for water filling/cooldown operations of the TSC and its contents. No confinement credit is taken by the quick-disconnect valved nipples during storage operations.

Following closure lid welding, hydrostatic pressure testing, draining, drying and backfilling of the TSC cavity with helium, the vent and drain openings were closed by welding in place SA240, Type 304/304L stainless steel, ½-inch thick x 4.4-inch diameter port cover plates to the closure lid vent and drain port recesses using a ¼-inch partial penetration weld. NDE of the port cover plate-to-closure lid welds was performed by PT examination of the final pass in accordance with the ASME Code, Section V, Article 6, with weld acceptance criteria per Section III, Subsection NB, Article NB-5350. After welding of the inner port cover plates in the vent and drain recesses, the confinement boundary of the inner port cover plates was tested for helium leakage to leak-tight criteria in accordance with ANSI N14.5 [3.9.26] requirements. Following successful helium leakage testing of the inner port cover plates, the outer port cover plates were welded to the closure lid. The final TSC closure operation was the installation and welding of the closure ring over the closure lid weld. The closure ring was welded to the TSC shell and closure lid to provide a secondary confinement boundary using ¼-inch partial penetration welds with final surface PT examination in accordance with the ASME Code, Section V, Article 6, with weld acceptance criteria per Section III, Subsection NB, Article NB-5350. The TSC closure lid is provided with six 1 ½-6 UNC-2B threaded holes for engagement of lifting hoist rings or other handling components and are designed for the single-failure-proof handling of the loaded and closed TSC. Optional stainless steel threaded plugs may be installed flush in the closure lid threaded holes to minimize radiation streaming effects during storage.

Each TSC assembly includes a basket structure that corresponds to the length and fuel assembly type/class. The fuel basket structure positions and supports the fuel assemblies in a subcritical array based on physical spacing, neutron absorbing poison materials, and in the case of PWR fuel assembly baskets, the use of a “flux trap” between adjacent assemblies. All fuel baskets have been designed and fabricated in accordance with the requirements of the ASME Code, Section III, Division 1, Subsection NG, to the maximum practical extent, with NRC approved exemptions.

Each fuel basket is an assembled structure of SA593, Type 630 17-4pH stainless steel support disks and SA240, Type 304 stainless steel top and bottom weldments installed on eight tie rods (YR-MPC or MPC-LACBWR) or six tie rods (CY-MPC) and aluminum heat rejection disks are interspersed with the 17-4pH stainless steel support disks in an alternating pattern. The PWR fuel basket assembly is a right-circular configuration with either twenty-four (24) or twenty-six (26) square fuel tube openings (CY-MPC) or thirty-six (36) (YR-MPC) laterally supported by the support disks and weldments, and axially restrained by the top and bottom weldments. The MPC-LACBWR BWR fuel basket assembly is a right-circular configuration with sixty-eight (68) square fuel tube openings laterally supported by the support disks and weldments, and axially restrained by the top and bottom weldments. The basket is assembled on eight (i.e., for YR-MPC and MPC-LACBWR) or six (i.e., CY-MPC) 1 ⅝-inch-diameter tie-rods fabricated from SA479, Type 304 stainless steel bar. The ½-inch (i.e., YR-MPC and CY-MPC) or 1-inch-thick (i.e., MPC-LACBWR) bottom weldment, fabricated from SA240, Type 304 stainless steel, is installed on six or eight tie rods and is positioned axially by six or eight, SA479/SA240, Type 304 stainless steel, 2-inch-thick, 3-inch-diameter support pads that are welded to the base of the bottom weldment. Additionally, SA240/SA479, Type 304 stainless steel ½ or ¾-inch-thick by 1-½ or 1-inch-high supports are welded to the base of the bottom weldment to axially position the basket assembly off the bottom of the TSC to facilitate the draining of cavity water. The bottom weldment supports interface with the four TSC location lugs, which maintain basket rotational alignment and structurally reinforce the bottom weldment.

The fuel baskets were assembled with the alternate installation of support disks and aluminum heat transfer disks positioned using stainless steel spacers, split spacers and washers that position the ½-inch-thick Type 6061-T651 aluminum alloy heat transfer disks between each support disk. The total number of support disks and aluminum disks varies based on the design decay heat load and length of each fuel basket type. After installation of the top-most support disk, the specified fuel tubes were installed into the basket assembly. The A240, Type 304 stainless steel fuel tubes are sized to allow passage through the support and heat transfer disks, but the tube is restrained by the bottom weldment that has smaller machined openings. Each fuel tube has none, one, two, three, or four sheets of neutron absorber depending on the fuel type held in place on the exterior of the tube by stainless steel sheathing (A240, Type 304). The eight top spacers were then used to position the 1-inch-thick (i.e., YR-MPC and MPC-LACBWR) or ½-inch thick (i.e., CY-MPC) SA240, Type 304 stainless steel top weldment. The top weldment is reinforced by a SA240, Type 304 stainless steel ring and stainless-steel support plates and shield baffles. The top weldment is held in place by SA479, Type 304 stainless steel top nuts, fabricated from 3-½-inch bar, that are installed on the eight tie rods. Following torquing, the top nuts were welded to the top weldment to prevent loosening.

3.1.1.2 Vertical Concrete Cask (VCC)

The NAC-MPC System VCC is the storage overpack for the TSC. The VCC can be provided in three different heights to accommodate the three NAC-MPC System TSC designs. The VCC assembly is constructed primarily from steel-reinforced concrete and carbon steel. The main wall component of the VCC assembly is constructed from normal weight concrete (e.g., minimum density of 140 pcf and compressive strength of 4,000 psi) made from Type 2 Portland cement and reinforced with number 6 ASTM A615/A615M carbon steel rebar. The internal cavity of the VCC assembly is lined by the 3-½ inch (YR-MPC and CY-MPC) or 2-½ inch (MPC-LACBWR) thick ASTM A36 carbon steel liner with a 2-inch-thick top flange and 2-½ x 3-inch shield ring (YR-MPC and CY-MPC only). The liner assembly rests on a 1-inch-thick base weldment fabricated from ASTM A36 carbon steel. The base weldment includes the bottom plate, four inlet vent assemblies and the baffle weldment. ASTM A36 carbon steel outlet vent assemblies are positioned below the shield ring and penetrate the upper concrete shell. The VCC annulus is closed by a shield plug assembly (YR-MPC and CY-MPC only) fabricated from 3-¾ inch and ⅜-inch-thick ASTM A36 carbon steel plates enclosing a layer of neutron shielding, either NS-3 or NS-4FR. The shield plug rests on the shield ring. The top closure of the VCC cavity is provided by the 1-½-inch-thick ASTM A36 carbon steel VCC lid (YR-MPC and CY-MPC only) bolted to the top lid by six stainless steel hex head bolts. The MPC-LACBWR VCC is closed by a 9.9-inch height composite steel enclosed concrete lid constructed of a 1.5-inch thick top A36 steel plate and a ⅜-inch thick bottom A36 steel plate encasing an 8.1-inch-thick layer of concrete. The single MPC-LACBWR VCC lid incorporates the function of both the shield plug and lid.

Exposed surfaces of the VCC carbon steel not covered by the concrete shell were coated with a two-part heat resistant coating such as Keeler & Long Kolor-Poxy Primer No. 3200 with a topcoat provided acrythane enamel Y-1 series topcoat, or equivalent. The NAC-MPC System VCC assembly also includes a Type 304 stainless steel sheet or on the top of the baffle weldment to support the loaded stainless steel TSC from contact with the carbon steel baffle surfaces. In addition, the YR-MPC VCC also includes a ⅛-inch thick layer of thermal insulation between the stainless-steel cover and the baffle weldment. At specific facilities, optional supplemental inlet vent shielding may be provided by either fixed or removable shield assemblies. The shields are provided by 4-inch diameter pipe, tubing or bar meeting ASTM A53 Gr. B or A106 Gr. B of pipe, A519 for tubing, or A36 for bar carbon steel. Inlet and outlet vents are closed by stainless steel screen assemblies retained by stainless steel washers and screws.

3.1.1.3 Transfer Cask (TFR) Assembly

The NAC-MPC System Transfer Casks (TFR) are special lifting devices designed, fabricated, tested, and maintained to meet the requirements of NUREG-0612 [3.9.24] and ANSI N14.6 [3.9.25]. Two separate TFRs were used for the three NAC-MPC System ISFSI projects with the main difference in height to allow acceptance of the three lengths of NAC-MPC System TSCs.

The YR-MPC TFR was used for the TSC transfer operations at both YR and DPC's LACBWR facility, and the CY-MPC TFRs were used exclusively at the CY facility.

The TFR main body assembly materials of construction consist of primarily ASTM A588 low alloy steel (e.g., inner and outer shells, bottom plate, top plate, retaining ring, shield door neutron shield boundary and male connector, trunnion cap). The ¾-inch thick inner radial shell, 1-inch outer radial shell, and 2.0-inch-thick top and 1-inch-thick bottom plates form an annulus into which the approximately 3-½-inch-thick (YR-MPC TFR) or 4.0-inch-thick (CY-MPC TFR) lead gamma shield bricks (ASTM A20) are assembled and interlocked. NS-4-FR neutron shielding material was then poured in place to form a 2.0-inch-thick (YR-MPC TFR) or 2.75-inch-thick (CY-MPC TFR) layer before final closure of the cavity. Additional TFR components are constructed of ASTM A350 LF2 low alloy steel (e.g., 9.5-inch-thick shield doors, door rails, lifting trunnions). The door rails were welded to the lower plate of the main body and support the two shield doors. The two 10-inch diameter lifting trunnions penetrate through the inner and outer shells near the top of the cask body and were welded to the inner and outer shells. The TFR also features an ASTM A588 low alloy steel ¾-inch thick retaining ring, bolted to the upper plate by twenty-four ASTM A193, Gr. B6 bolts which prevents the TSC from being accidently removed from the TFR annulus during the loaded TSC transfer operation. In order to ensure that the shield doors remained closed during lifting and handling of the TFR, door lock pin assemblies are installed on both sides of the bottom plate for each shield door. During operations at least one of the two lock bolts is required to be installed for each door assembly. All exposed air-facing carbon steel surfaces of the NAC-MPC System TFRs and their subcomponents, except those noted below, are coated with Carboline 890 or Keeler & Long E-series epoxy enamel or equivalent spent fuel compatible coating system. The coating was to protect the spent fuel chemistry during in-pool operations, facilitate TFR decontamination, and provide corrosion protection for TFR surfaces. To prevent paint removal in the area of the trunnions, stainless steel scuff plates are welded to the outer shell. The only exposed carbon steel TFR components that are not required to be coated are the door rails and interfacing mating surfaces of the shield doors which are coated with a spent fuel compatible lubricant such as Neolube or equivalent to facilitate operation using the hydraulic cylinders installed on the interfacing transfer adapter plate. A total of ten penetrations (two upper and eight lower inlets/outlets) are provided through the TFR body using ASTM A312 stainless steel pipe. The inlet/outlet penetrations are used to provide filtered pool water to minimize the contamination of the TSC exterior surfaces by limiting contact with the contaminated spent fuel pool water.

Each NAC-MPC System TFR was provided with a Transfer Adapter Plate designed to rest on the top of the VCC as an interface device with the TFR. The main functions of the Transfer Adapter are to engage the TFR door connectors to mating connectors to allow the doors to be opened by hydraulic cylinders when the doors are unlocked, and to provide additional side shielding to protect plant personnel during actual lifting and lowering of the TSC from the VCC.

3.1.1.4 Spent Nuclear Fuel (SNF) Assembly

The SNF assembly subcomponents consist of stainless-steel or zircaloy fuel rod cladding, zircaloy or stainless-steel spacer grids and guide tubes or water tubes, and stainless steel and/or Inconel top and bottom end nozzle structures. BWR SNF assembly fuel rods may have partial length neutron absorbing materials. PWR SNF assemblies may also include various assembly control components, such as burnable poison rod assemblies, thimble plug assemblies, and control rod assemblies. The insert materials include zircaloy or stainless-steel cladding, stainless steel or Inconel top fittings, and neutron absorbing materials such as boron carbide, borosilicate glass or silver-indium-cadmium. SNF assemblies may also contain zircaloy or stainless-steel dummy rods in place of fuel rods in one or more array locations.

3.1.2 Environments

3.1.2.1 NAC-MPC System Operating Site Environments

The second step in the aging management review process is the identification of the specific operating environments for each of the SSC subcomponents that are ITS. The potential operating environments for the NAC-MPC System are discussed in this section. With the exception of the SSC subcomponents that are exposed to the helium (inert gas) atmosphere within the TSC cavity and the fully encased in steel (air-sealed) environments between the shield lid and structural lid, shield plug and quick disconnect fittings of the TSC (YR-MPC and CY-MPC, only), the fully encased (neutron shield/lead) in steel cavity between the inner and outer shells of the TFR, and the fully encased in steel of the neutron shielding materials in the shield plug (YR-MPC and CY-MPC, only), the environment to which each subcomponent of the in-scope SSC is exposed depends on the characteristics of the facility site environment and their location within the system.

NAC-MPC Systems are currently deployed at three nuclear plant sites: the Yankee Rowe decommissioned site in Rowe, Massachusetts adjacent to the Sherman Reservoir; the Connecticut Yankee decommissioned site in Haddam Neck, Connecticut located adjacent to the Connecticut River, and Dairyland Power Cooperative' decommissioned LACBWR site in LaCrosse, Wisconsin located adjacent to the Mississippi River. None of the sites is located near a marine environment or utilized cooling tower systems during plant operation. All three sites are located above the freeze line in the northern US and experience low winter temperature and conditions, and moderate levels of rainfall and humidity. All three of the sites fall within the evaluated environmental conditions evaluated in the NAC-MPC FSAR [3.9.1.a - 3.9.1.m]. The 30-year average monthly temperatures range from approximately 22.6°F in January to 69.2°F in July at YR, 26.9°F to 72.6°F at CY, and 17.4°F to 73.7°F at LACBWR. (Temperature data obtained from NOAA and are average monthly high and low temperatures for the period from 1981 thru 2010).

3.1.2.2 Specific Environments Identified for NAC-MPC Systems

There are six basic types of environments identified that envelope the conditions of the MPC SSC subcomponents as discussed below: Helium; Fully Encased (Steel); Sheltered; Embedded (Concrete); Air-Indoor/Outdoor; and Air-Outdoor.

3.1.2.2.1 Helium (HE) - TSC Cavity Inert Gas

The SNF assemblies, fuel basket assembly, and the inside (cavity facing) surfaces of the TSC shell assembly, and shield/closure lid are all exposed to the helium environment inside the TSC cavity. The average temperature of this gas can range from the ambient air temperature for a zero-decay heat load to a maximum of 367°F for the maximum CY-MPC canister heat load of 17.5 kW. (Note: YR-MPC maximum heat load is 12.5 kW and MPC-LACBWR maximum heat load is 4.5 kW). The gas pressure in the TSC cavity is close to one atmosphere with a calculated maximum normal operating pressure of approximately 9 psig. The presence of moisture, oxygen or oxygen generating gases is limited to very low levels by the vacuum drying process and final cavity evacuation to ≤ 3 torr prior to final helium backfill to preclude deleterious chemical changes or degradation of the fuel cladding. In addition to the elevated temperatures and trace amounts of oxygen and/or moisture, the TSC interior components are exposed to significant gamma and neutron radiation.

3.1.2.2.2 Fully Encased (FE) - Steel

The fully encased environment applies for materials that are fully enclosed inside another component or fully lined by another material (e.g., steel), which prevents ingress of water and contaminates.

In the NAC-MPC System the fully encased in steel environments include the NS-3 or NS-4-FR poured in the VCC shield plug (YR-MPC and CY-MPC, only), which is fully encased in a steel plate enclosure. In addition, the NS-4-FR and lead gamma shield bricks of the NAC-MPC TFR assembly are fully encased inside the enclosure formed by the inner and outer steel shells and top and bottom steel plates. The primary issue for fully encased in metal environments is any potential for chemical reactions between the two or more materials meeting at a given surface. Any such reactions will be potentially governed by temperature and the associated chemistry of the combination of the embedded materials. Temperatures of the embedded NS-3/NS-4-FR in the VCC shield plug could range from ambient to as high as 160°F for maximum decay heat load of 17.5 kW (CY-MPC) and 100°F full solar conditions. TFR assembly embedded materials may be exposed to elevated temperatures (250°F) for short durations during fuel loading, transfer and unloading operations. During storage, the TFR assembly temperatures will be maintained within a narrow range of “room temperature” when stored in a building or normal outside ambient conditions if stored outside of the facility. The radiation levels of the fully encased in metal components discussed above are significantly lower than those experienced by the sheltered air environment.

In addition, for the CY-MPC and YR-MPC TSCs following the completion of the welding of the shield lid to the TSC shell, the TSC cavity draining is completed, and vacuum drying, and helium leakage testing operations are performed. The structural lid is installed and welded to the TSC shell completing the closure of the TSC. The small free volumes that exist between the structural lid and the top of the shield lid, and the port covers, and the ports valved recesses, are filled with ambient air from inside of the building in which the TSC closure operations were performed and are considered as a fully encased in metal environment. The temperature of this sealed air environment during storage operations may range from ambient air-outdoor temperatures for zero decay heat to a maximum of approximately 199°F for the design basis decay heat load of 17.5 kW and steady state severe hot ambient temperature conditions. The small volume of ambient indoor air that is sealed in the free volume between these subcomponents may initially contain a limited amount of oxygen. Unlike the sheltered environment, the sealed air will not be replenished, and therefore, the amount of potential corrosion that can occur to the stainless-steel surfaces exposed to this environment is limited by the small amount of oxygen initially present in the free volume. Therefore, the corrosion resistance of the stainless-steel materials and limited free oxygen in the free space ensure that corrosion of these surfaces exposed to this environment is insignificant and does not affect the intended safety functions of these subcomponents.

3.1.2.2.3 Sheltered Environment (SH)

The outer surfaces of the TSC assembly and the interior surfaces and components of the VCC assembly (inner surfaces of the liner shell, liner base weldment and baffle weldment, inlet and outlet assemblies, top side of the baffle coverplate, underside of the VCC lid, and all surfaces of the shield ring and shield plug) are exposed to a sheltered environment (SH). This environment includes ambient air, but not sun, rain, or wind exposure. The ambient air may contain moisture and some salinity. The temperature of the ambient air inside the VCC cavity may range from that of the outside air for zero decay heat to nearly 310°F based on the peak temperature of the TSC shell of 312°F for the design-basis heat load of 17.5 kW and extreme hot off-normal ambient conditions. Generally, the elevated temperatures of the sheltered environment air will keep moisture levels below those seen on the outer surfaces of the NAC-MPC VCC. Components exposed to the sheltered environment experience reduced levels of gamma and neutron radiation than those seen in the TSC interior environment.

3.1.2.2.4 Embedded (Concrete) Environment (E-C)

The embedded environment applies for materials that are in contact with another material or component. This may prevent ingress of water and contaminants to the embedded surface, depending on the permeability of the embedding environment.

These embedded in concrete environments include the metal components of the NAC-MPC VCC assembly that are either cast inside or against concrete, such as the outer surfaces of the liner shell, top of the VCC base plate, underside of the liner top flange, concrete-side facing surfaces of the inlet and outlet vent structure, and the reinforcing rebar embedded in the concrete shell.

The primary issue for embedded concrete environments is any potential for chemical reactions between the two or more materials meeting at a given surface. Any such reactions will be potentially governed by temperature and the associated chemistry of the combination of embedded materials. For the VCC assembly the primary issue is any potential reaction between carbon steel and concrete. The temperature of the VCC embedded materials at the concrete to carbon steel interface could range from ambient temperature to as high as 171°F for a decay heat load of 17.5 kW.

3.1.2.2.5 Air-Outdoor Environment (OD)

During NAC-MPC System storage operations, all exterior surfaces of VCC are exposed to all weather-related effects, including insolation, wind, rain/snow/ice (possibly containing salts), and ambient air at the plant site. The steel plate that forms the bottom surface of the VCC base weldment assembly is also exposed to water and potential icing, as it is in direct contact with the ISFSI pad but is sheltered from sun and wind. The ambient temperature for normal and extreme weather conditions range from -40°F to 125°F. The moisture and salinity levels to which the exterior surfaces of the VCC assembly are exposed may vary widely for the three NAC-MPC System ISFSIs, although none of the sites is in a high salinity marine environment. The radiation levels on the exterior surfaces of the VCC assembly are sufficiently low to satisfy the applicable Technical Specification dose rate limits.

3.1.2.2.6 Air-Indoor/Outdoor Environment (OD)

The air-indoor/outdoor environment applies to the NAC-MPC System Transfer Cask (TFR) components that are typically housed indoors except for periodic exposure to outdoor air during TSC transfer operations. Indoor air describes the environment in a spent fuel building or other protective enclosure. At NAC-MPC System ISFSIs that have completed NAC-MPC loading operations, TFR components are stored outdoors in a storage container or covered by a protective covering.

Following completion of NAC-MPC fuel loading operations, the current TFR assemblies are stored outside with limited protection from environmental extremes. Stored TFR assemblies are not exposed to the elevated temperatures and radiation levels experienced by the TSC and VCC during storage operations except for the short durations of the cask system loading, handling and unloading operations. Also, the interior and exterior surfaces of the TFR assembly are fully accessible for inspection and repair whereas the TSC assembly exterior and VCC assembly interior surfaces are not routinely accessible.

For purposes of the evaluation of aging effects in different environments, the air-indoor/air-outdoor environment is evaluated under the air-outdoor environment as no component is exposed exclusively to an air-indoor environment.

3.2 IDENTIFICATION OF AGING EFFECTS REQUIRING MANAGEMENT

The third step in the aging management review process involves the identification of the aging effects requiring management. Aging effects requiring management during the period of extended operation are those that could cause a loss of passive SSC and SSC subcomponents intended functions. If the degradation of SSC subcomponents would be insufficient to cause a loss of function, or the relevant conditions do not exist at locations that utilize the NAC-MPC System for the aging effect to occur and propagate, then aging management is not required.

Potential aging effects, presented in terms of material and environmental combinations, have been evaluated and those aging effects requiring management have been determined and identified in this application. Both potential aging effects that theoretically could occur, as well as aging effects that have occurred based upon industry and NAC-MPC System user operating experience, were considered. The evaluation was applied to identified SSC subcomponents. A summary table of the SSC subcomponent materials versus the operating environments and aging effects requiring aging management is provided in Table 3.2-10.

The environments considered in this evaluation are the environments that the SSC subcomponents normally experience. Environmental stressors that are conditions not normally experienced (such as accident conditions), or that may be caused by a design problem, are considered event-driven situations and have not been characterized as sources of aging. Such event-driven situations would be evaluated and subsequent corrective actions, if any, implemented at the time of the event.

Aging effects are the manifestation of aging mechanisms. To effectively manage an aging effect, it is necessary to determine the aging mechanisms that potentially affect a given material under certain environmental conditions. Therefore, the aging management review process identifies both the aging effects and the associated aging mechanisms which cause them. Various mechanisms are only applicable under certain conditions, such as high temperature or moisture, for example. Each identified mechanism was characterized by a set of applicable conditions that must be met for the mechanisms to occur and/or propagate. Given this evaluation process, 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.

Aging effects, and the mechanisms that cause them, are evaluated for the combinations of materials and environments identified for the subcomponent of the in-scope SSC based upon a comprehensive review of known literature, industry operating experience, and maintenance and inspection records. Possible or theoretical aging effects for the materials of construction used in the NAC-MPC System are determined primarily from research of literature of degradation mechanisms including the following:

- NUREG-1927, Revision 2, Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel [3.9.2]
- NEI. NEI 14-03, Revision 2, “Guidance for Operations-Based Aging Management for Dry Cask Storage,” December 2016. [3.9.3]

- NUREG-2214, Final Report, July 2019, Managing Aging Processes in Storage (MAPS) Report [3.9.4]
- American Society for Testing and Materials (ASTM) C 1562-03 [3.9.5]
- Electric Power Research Institute (EPRI) Report TR-1003416 [3.9.6]
- EPRI Technical Report TR-108757 [3.9.7]
- EPRI Technical Report TR-1002882 [3.9.8]
- International Atomic Energy Agency Technical Report Series No. 443 [3.9.9]
- NRC Interim Staff Guidance (ISG) 11, Revision 3 [3.9.10]
- NUREG/CR-6745, Dry Cask Storage Characterization Project [3.9.11]
- NUREG/CR-6831, Examination of PWR Fuel Rods after 15 Years in Dry Storage [3.9.12]
- NUREG-1522, Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures [3.9.13]
- NUREG-1801, R2, Generic Aging Lessons Learned (GALL) Report [3.9.14]
- EPRI Technical Report, TR-3002005371, Susceptibility Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless-Steel Canisters for Dry Storage [3.9.15]
- EPRI Technical Report, TR-3002008193, Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless-Steel Canisters [3.9.16]
- EPRI Technical Report Update. TR-3002002785, Failure Modes and Effects Analyses (FEMA) of Welded Stainless Steel Dry Cask Storage Canisters [3.9.17]
- NRC Interim Staff Guidance (ISG) -2, Revision 2, Fuel Retrievability in Spent Fuel Storage Applications [3.9.18]
- NUREG/CR-7170, Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Chloride and Non-Chloride Salts [3.9.19]
- NRC Report, Identification and Prioritization of the Technical Information Needs Affecting Potential Regulation of Extended Storage and Transport of Spent Nuclear Fuel [3.9.20]
- DOE/ANL Report ANL-12/29 “Managing Aging Effects on Dry Cask Storage Systems for Extended Long-Term Storage and Transportation”, 2012 [3.9.21]
- NRC Information Notice 2011-20, Concrete Degradation by Alkali-Silica Reaction [3.9.22]
- NRC Interim Staff Guidance (ISG) -24, Revision 0, The Use of a Demonstration Program as a Surveillance Tool for Confirmation of Integrity for Continued Storage of High Burnup Fuel Beyond 20 Years [3.9.23]

Aging effects that have occurred during the initial storage period for the NAC-MPC System are determined based on a review of the available licensee records and operating experience. Aging effects that could adversely affect the ability of the in-scope SSC to perform their safety function(s) require additional Aging Management Activity (AMA) to address potential degradation that may occur during the period of extended operation. These additional AMAs consist of either Time-Limited Aging Analysis (TLAA) or Aging Management Programs (AMPs), as discussed in Section 3.3 and 3.4, respectively. The possible and observed aging effects and associated aging mechanisms identified for the in-scope SSC for the period of extended operation are discussed in the following sections and summarized in Tables 3.2-1 through Table 3.2-9. The tables address each individual NAC-MPC System SSCs (e.g., YR-MPC TSC, CY-MPC TSC, etc.) as each individual system has different sets of License Drawings and minor differences in components and identification. The description of the aging effects and mechanisms on the materials of SSCs and subcomponents that are ITS are extracted from data provided in Section 3 of the MAPS report [3.9.4] and contained MAPS references are provided in parentheses.

3.2.1 Casks and Internals

Casks and internals include various metallic subcomponents of the Vertical Concrete Cask (VCC), the Transportable Storage Canister (TSC) or canister, the fuel baskets, and other internal subcomponents, the Transfer Cask (TFR) and Transfer Adapter. The NAC-MPC System VCC, TSC, and fuel basket assembly and internal subcomponents, and TFR/Transfer Adapter contain various metallic subcomponents that are exposed to several environments within and outside the system such as sheltered environments, indoor-outdoor air, outdoor air, helium, and fully encased environments. The spent nuclear fuel (SNF) also exposes subcomponents to elevated temperatures and radiation, with heat exposure and dose depending on the subcomponent location and the SNF characteristics (e.g., burnup and age of fuel). The materials of construction for these subcomponents include steel, stainless steel, aluminum alloy, and lead.

A set of known aging mechanisms for metallic cask and internal subcomponents was established by the NRC in MAPS [3.9.4] including environmental, thermal, mechanical, and irradiation-induced aging mechanisms as follows:

- general corrosion
- pitting and crevice corrosion
- galvanic corrosion
- Microbial Induced Corrosion (MIC)
- Stress Corrosion Cracking (SCC) (including hydrogen embrittlement)
- creep
- fatigue
- thermal aging
- radiation embrittlement
- stress relaxation
- wear

Not all these mechanisms are credible for each structure, system, and component (SSC) of the NAC-MPC System. For example, temperatures are not considered sufficiently high to cause creep of steel and stainless-steel subcomponents. Also, general corrosion is not considered to be a credible aging mechanism for subcomponents fabricated from stainless steels, because these materials exhibit passive behavior and negligible general corrosion rates. Detailed discussions regarding potential aging mechanisms for each NAC-MPC System SSC subcomponent material and the technical bases for those requiring aging management are detailed in the following subsections.

3.2.1.1 Steel (Carbon, Low-Alloy, High-Strength Low-Alloy)

In the NAC-MPC System steel subcomponents are used in the VCC and TFR/Transfer Adapter SSCs and are exposed to sheltered, outdoor air, indoor-outdoor air, and embedded in concrete environments. The exterior surfaces of NAC-MPC System VCC steel subcomponents are coated with epoxy or inorganic zinc to mitigate corrosion; however, these coatings can degrade, resulting in exposure of steel to the atmosphere. Steels used for the NAC-MPC System TFRs/Transfer Adapters are predominately exposed to an indoor environment, except for short periods of outdoor exposure during transfer operations. For such air-indoor/outdoor environment exposure, aging effects from aqueous corrosion processes are expected to be bounded by the outdoor environment. As such, the indoor air environment is not discussed separately.

3.2.1.1.1 General Corrosion

General corrosion, also known as uniform corrosion, proceeds at approximately the same rate over a metal surface and freely exposed steel surfaces in contact with moist air or water are subject to general corrosion. The corrosion rate depends on solution composition, pH, and temperature.

Steel Subcomponents Exposed to Outdoor and Sheltered Environments

In outdoor conditions, rain, fog, snow, and dew condensation can generate moisture layers on the steel surface that cause general corrosion. Atmospheric corrosion rates can vary from 0 to 0.2 millimeters/year (mm/yr.) [0 to 7.9 mils/yr.] depending on relative humidity, temperature, and levels of chloride and pollutants in the atmosphere [3.9.117].

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 non-chloride-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 humidities in nature [3.9.19]. The heat generated by the radioactive decay of spent fuel decreases over time. VCC steel subcomponents exposed to sheltered environments are located farther away from the fuel compared to the stainless-steel canister shell and 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, general corrosion is considered to be credible, and therefore, aging management is required during the 40-year period of extended operation. The applicable AMPs proposed to evaluate this aging mechanism are the External VCC Metal Components Surface Monitoring AMP and the Transfer Cask AMP and are discussed in Section 3.4. The potential for general corrosion of the VCC internal steel components (e.g., liner, pedestal, baseplate and inlets/outlets) is evaluated in a TLAA for the 40-year period of extended operation as discussed in Section 3.3.

Steel Components Exposed to Demineralized Water

Except for short durations of immersion of the NAC-MPC System TFR in the spent fuel pool, there are no steel components of the NAC-MPC System exposed long-term to demineralized water.

The NAC-MPC System TFR carbon steel components are coated with spent fuel pool compatible coating systems that are maintained as part of the TFR maintenance program and the cask is deconned and dried after each in-pool immersion. Therefore, the environment defined as steel components exposed to demineralized water is not included in the evaluation of aging mechanisms requiring aging management, and no aging management activities except normal TFR coating maintenance have been identified as required.

Steel Subcomponents Exposed to Groundwater or Soil

There are no NAC-MPC System steel components exposed to groundwater or soil, and therefore, aging management review for this environment is not required.

Steel Subcomponents Exposed to an Embedded (Concrete) Environment

In the NAC-MPC System VCC, steel rebar, nelson studs and other subcomponents are embedded in the concrete shell and the concrete is in contact with outdoor air. When the VCC concrete shell is intact, the alkaline concrete solution passivates the steel. Embedded steel components including rebar, Nelson studs, and SSCs partially embedded in concrete such as the outer surface of the steel liner, baseplate, top flange, have been evaluated for general corrosion aging effects in a TLAA which concluded that these components will maintain the capability to perform their intended safety functions for the 40-year PEO as discussed in Section 3.3.

Steel Subcomponents Exposed to a Fully Encased (Steel) Environment

In the NAC-MPC System, polymer-based or cement-based neutron-shielding materials are poured into the VCC shield plug, and polymer-based neutron shielding is poured between the TFR outer shell and lead bricks/inner shell, leaving one side of the steel encased. The neutron-shielding materials include NS-4-FR or BISCO NS-3. Because of the encased steel has limited exposure to water and oxygen, general corrosion is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

Steel Subcomponents Exposed to Helium

In the NAC-MPC System, there are no steel subcomponents exposed to a helium environment, as all NAC-MPC System TSC and fuel basket steel components are stainless steel. Therefore, aging management of steel in a helium environment for general corrosion is not required for the NAC-MPC System during the 40-year period of extended operation.

3.2.1.1.2 Pitting and Crevice Corrosion

Pitting corrosion is a localized form of corrosion that is confined to a point or small area of a metal surface [3.9.75]. It takes the form of cavities called pits. Crevice corrosion is another localized form of corrosion that occurs in a wetted environment when a crevice exists [3.9.97]. It occurs more frequently in connections, lap joints, splice plates, bolt threads, under bolt heads, or at points of contact between metals and nonmetals. Crevice corrosion is associated with stagnant or low-flow solutions. As discussed previously, the common form of corrosion for steel is general corrosion. However, steel is also known to be susceptible to pitting and crevice corrosion in an oxidizing and alkaline environment, especially in the presence of chlorides. The exterior surfaces of some subcomponents are coated with epoxy or inorganic zinc to mitigate corrosion (e.g., the external surfaces of the NAC-MPC System TFR/Transfer Adapter and VCC steel surfaces exposed to outdoor air or sheltered). Depending on the quality and chemical composition of the coating, water and corrosive agents can permeate coating defects, initiating pitting. After initiation of a coating defect, the coating could function as a crevice former and initiate crevice corrosion.

Steel Subcomponents Exposed to Air-Outdoor and Sheltered Environments, and Embedded (Concrete) Environments

The potential to form aqueous electrolytes on surfaces exposed to outdoor and sheltered environments is present, 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, the steel components can be exposed to water containing dissolved carbonates and chlorides, which could be conducive to pitting and crevice corrosion as well.

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 [3.9.136]. For external VCC components steel components exposed to outdoor air (external surfaces of VCC lid.) are evaluated by TLAA 30013-2003 for external and internal carbon steel VCC components and were shown to maintain their safety margins for the 40-year PEO as discussed in Section 3.3. For this aging mechanism, the Transfer Cask and Transfer Adapter surfaces exposed to outdoor air are evaluated for pitting and crevice corrosion aging mechanisms. The AMP for the monitoring of these components is the Transfer Cask and Transfer Adapter AMP as discussed in Section 3.4

Therefore, aging management of certain steel exposed to E-C environments such as reinforcing rebar is required during the 40-year period of extended operation. The potential for pitting and crevice corrosion of the VCC internal steel components (e.g., liner, pedestal, baseplate and inlets/outlets) and steel components embedded in concrete (e.g., reinforcing rebar, Nelson studs) and partially embedded components (e.g., liner, top flange, inlet and outlet top and side plates, baseplate, stand and support plates) were evaluated in TLAA 30013-2003 and were shown to maintain their safety margins for the 40-year PEO as discussed in Section 3.3.

Steel Subcomponents Exposed to Fully Encased (Steel) Environments

In the NAC-MPC System, polymer-based or cement-based neutron-shielding materials are poured into the VCC shield plug, and polymer-based neutron shielding is poured between the TFR outer shell and lead bricks/inner shell, leaving one side of the steel embedded. The neutron-shielding materials include NS-4-FR or BISCO NS-3. Because the fully encasing steel side plates of the neutron-shielding materials has no exposure to water and oxygen, pitting and crevice corrosion of the steel is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

Steel Subcomponents Exposed to Helium

In the NAC-MPC System, there are no steel subcomponents exposed to a helium environment, as all NAC-MPC System TSC and fuel basket steel components are stainless steel. Therefore, aging management of steel in a helium environment for pitting and crevice corrosion is not required for the NAC-MPC System during the 40-year period of extended operation.

3.2.1.1.3 Galvanic Corrosion

Galvanic corrosion occurs when two dissimilar metals or conductive materials are in physical contact in the presence of a conducting solution [3.9.37; 3.9.84]. Under these conditions, an electrolytic cell is formed, transmitting an electrical current between an anode and a cathode. Oxidation occurs at the anode, and reduction occurs at the cathode. The extent of galvanic corrosion depends on potential differences between the two metals, surface area ratio of the anode and cathode, environment, reaction kinetics, corrosion products, and other factors [3.9.37]. In general storage systems, galvanic coupling can exist between steel and other more noble materials such as stainless steel, graphite, nickel, and brass. These galvanic couples can be exposed to sheltered and outdoor air environments.

Steel Subcomponents Exposed to Outdoor and Sheltered Environments

Aqueous electrolytes for subcomponents exposed to outdoor and sheltered environments are present during the 40-year period of extended operation. In the NAC-MPC System, there is a direct connection between SSC subcomponent steel and more noble materials such as stainless steel. The points of connection are in the VCC and TSC are between the bottom of the TSC, the ¼-inch stainless steel cover plate or the stainless steel coverplate and a ⅛ inch layer of silicone insulation (YR-MPC only) and the top of the VCC baffle weldment base plate. However, the potential for galvanic corrosion of the TSC stainless steel bottom plate is precluded by the presence of a ¼-inch-thick stainless-steel cover plate. The potential for significant corrosion of the epoxy coated or inorganic zinc VCC baffle weldment is limited due to the thickness of the baffle weldment top plate (2 inch).

Other points of connection between SSC steel subcomponents and more noble materials such as stainless steel or high strength steel threaded into carbon steel components. This includes the VCC stainless steel bolts and washers that attach the coated VCC lid to the coated VCC flange and the high strength stainless steel retaining ring bolts that attach the coated Transfer Cask retaining ring to the top flange. The threaded components are inspected every 5 years in accordance with AMP-3, Aging Management Program for External Vertical Concrete Casks (VCC)-Metallic Components Monitoring. The VCC lid external surfaces are also inspected for any indications of loss of material due to corrosion. The AMP-3 inspection program includes inspection of all external metallic VCC components including the VCC lid bolts and washers, and VCC lid for any indication of obvious loss of base material and visual evidence of loose or missing bolts, physical displacement, and other conditions indicative of loss of preload on VCC lid.

In the case of TFR retaining ring bolting, the bolting and retaining ring are only installed during TSC transfer operations from the TFR to the VCC and are never immersed in the spent fuel pool. After each operation, the bolts and retaining ring are removed, decontaminated, and stored to the next transfer sequence. The TFR and its components are inspected in accordance with ANSI N14.6 during loading operations on a quarterly and annual basis. In addition, AMP-5 Aging Management Program for Transfer Casks (TFR) and Transfer Adapters requires a periodic inspection of TFRs and Transfer Adapters on a five-year interval or prior to the next use of the TFR/Transfer Adapter following a period of non-use.

Required aging management is provided by the proposed AMP-3 for the VCC components involved and AMP-5 for the TFR components involved.

3.2.1.1.4 Microbiologically Influenced Corrosion (MIC)

MIC is corrosion caused or promoted by the metabolic activity of microorganisms and active microbial metabolism that requires water in the form of water vapor, condensation, or deliquescence, and available nutrients to support microbial activity [3.9.58]. Biofilms can form even under radiation environments [3.9.56]. Bacteria resistant to radiation include *Micrococcus radiodurans*, which can tolerate 10 kilograys (kGy) [10^6 rads] of irradiation. MIC is limited where relative humidity is below 90 percent and negligible for relative humidity below 60 percent [3.9.99]. MIC has been found to be operable within a temperature range of −5°C to 110°C [23 to 230°F].

Although most of the evidence of MIC for metallic components is from conditions under which the metal surface is kept continuously wet, microorganisms can live in many environments, such as water, soil, and air, where aerobic bacteria (e.g., iron-manganese oxidizing bacteria, sulfur/sulfide oxidizing bacteria, methane producers, organic acid-producing bacteria), fungi, and algae can develop.

Steel Subcomponents Exposed to Groundwater/Soil and Embedded (Concrete) Environments

In the NAC-MPC System, steel SSC subcomponents (e.g., rebar, nelson studs, etc.) are embedded in the VCC concrete shell. However, the concrete surfaces are not exposed to groundwater or soil, and therefore, propagation of MIC in the VCC concrete shell is not expected to be significant. As such, MIC of steel in concrete environments is not considered to be credible for the NAC-MPC System, and therefore, aging management is not required during the 40-year period of extended operation. There are no NAC-MPC System steel components exposed to groundwater or soil, and therefore, aging management review for this environment is not required.

Steel Subcomponents Exposed to Sheltered and Air-Outdoor Environments

In the NAC-MPC System VCC steel components, the potential to form aqueous electrolytes for subcomponents exposed to outdoor and sheltered environments is present, either from direct exposure to precipitation or by deliquescence of deposited salts. These electrolytes have the potential to support microbial activity.

However, there is no operating experience of MIC degradation of steel engineering components that are exposed to environments similar to those of dry cask storage systems, where continuous exposure to a relative humidity above 90 percent is not expected. The operating experience of MIC for metallic components is largely from instances in which the metal surface was kept continuously wet. Because there is no applicable operating experience of MIC damage of steel under relevant atmospheric conditions, MIC is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

Steel Components Exposed to Demineralized Water

Except for short durations of immersion of the NAC-MPC System TFR in the spent fuel pool there are no steel components of the NAC-MPC system exposed long term to demineralized water as the NAC-MPC System TFR does not use demineralized water for neutron shielding. Therefore, these environments are not included in the evaluation of aging mechanisms requiring aging management.

Steel Subcomponents Exposed to Neutron-Shielding and Lead in a Fully Encased (FE) Steel Environment

In the NAC-MPC System, there are shielding materials fully encased (FE) in steel components in the TFR, VCC shield plug and VCC Lid Assembly (MPC-LACBWR only). However, due to the absence or limited amount of water and nutrients in the lead and neutron shield materials in the sealed air FE environments within the VCC shield plug, VCC Lid Assembly and TFR, MIC of steel is not credible for the 40-year period of extended operation, and therefore, aging management is not required.

3.2.1.1.5 Stress-Corrosion Cracking (SSC)

SCC is the cracking of a metal produced by the combined action of corrosion and tensile stress (applied or residual) [3.9.93]. SCC is highly chemical specific in that certain alloys are likely to undergo SCC only when exposed to a small number of chemical environments. SCC is the result of a combination of three factors: (1) a susceptible material, (2) exposure to a corrosive environment, and (3) tensile stresses. High-strength steels with yield strengths greater than or equal to 150,000 pounds per square inch (150 ksi) have been found to be susceptible to SCC under exposure to aqueous electrolytes [3.9.92; 3.9.112; 3.9.63].

Steel Subcomponents Exposed to Sheltered and Air-Outdoor Environments

In the NAC-MPC System high strength steel bolting of VCC subcomponents and the stainless steel TFR retaining ring bolting are torqued to low values and are below the stress threshold values required to initiate SCC. Because of the low applied stresses, SCC of steel bolts of the NAC-MPC System exposed to sheltered and air-outdoor environments is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.1.1.6 Creep

Creep is the time-dependent inelastic deformation that takes place at an elevated temperature and a constant stress [3.9.82]. Because the deformation processes that produce creep are thermally activated, the rate of this time-dependent deformation is a strong function of the temperature. The creep rate also depends on the applied stress but does not generally vary with the environment. As a general rule of thumb, at temperatures below $0.4T_m$, where T_m is the melting point of the metal in Kelvin (K), thermal activation is insufficient to produce significant creep [3.9.46]. Temperatures of at least 716 K (443°C [829°F]) are required to initiate creep in steels. However, the $0.4T_m$ rule of thumb underestimates the minimum creep temperature for steels, as temperatures above 500°C [932°F] have been found to be required for creep in steels [3.9.140]. The creep rate also depends on the applied stress but does not generally vary with the environment.

Steel Subcomponents Exposed to Helium

The highest temperatures within the NAC-MPC System are at locations close to the fuel rods. However, there are no steel components in the NAC-MPC System TSC and fuel basket, and therefore, are not applicable to this aging mechanism for the NAC-MPC System and aging management is not required during the 40-year period of extended operation.

Steel Subcomponents Exposed to Sheltered, Air-Outdoor, Embedded (all), and Fully Encased Environments

NAC-MPC System steel subcomponents in the VCC and TFR are exposed to sheltered, outdoor air, embedded (concrete), and fully encased steel environments. However, these subcomponents experience significantly lower temperatures than those experienced by the internal TSC subcomponents and are below the $0.4T_m$ threshold. Therefore, creep of these steel subcomponents is not considered to be credible, and aging management is not required during the 40-year period of extended operation.

3.2.1.1.7 Fatigue

Fatigue is the progressive structural damage that occurs when a metal is subjected to cyclic loading. Because spent fuel storage in a NAC-MPC System is a static application, cyclic loading by a purely mechanical means is largely limited to NAC-MPC System TFR lifting trunnions, which are loaded each time a TSC is moved from the spent fuel pool to placement in the VCC. Other subcomponents, however, could experience cyclic loads due to thermal effects.

Daily and seasonal fluctuations in the temperature of the external environment can impose stresses on materials as they expand, and contract while being constrained by adjacent components. The cyclic stress, σ, induced by these temperature fluctuations depends on many factors, including the material's coefficient of thermal expansion (α_0) and Young's modulus of elasticity (E), the actual change in temperature (ΔT), and the degree of constraint on the subcomponent

Due to the low temperatures of the NAC-MPC System steel components in the VCC and TFR, and limited cyclic stresses, fatigue is not expected to be a credible degradation method, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.1.1.8 Thermal Aging

The microstructures of most steels will change, given sufficient time at temperature, and this can affect mechanical properties. This process is commonly called thermal aging. The effect of thermal aging will depend on the time at temperature and the microstructure and carbon content of the steel subcomponents.

Steel Subcomponents Exposed to Helium

The highest temperatures within the NAC-MPC System are at locations close to the fuel rods. However, there are no steel components in the NAC-MPC System TSC and fuel basket, and therefore, these components are not applicable to this aging mechanism and aging management is not required during the 40-year period of extended operation.

Steel Subcomponents Exposed to Sheltered, Air-Outdoor, Fully Encased, and Embedded (Concrete) Environments

As stated above, undesired material property changes due to tempering of hardened steels could occur at temperatures greater than 200°C [392°F]. The temperatures of NAC-MPC System steel subcomponents of the VCC and TFR exposed to sheltered, outdoor air, embedded (concrete), and fully encased steel environments are bounded by the stainless steel TSC shell temperature, as these subcomponents are located farther away from the fuel. Time-temperature profiles calculated for the stainless-steel NAC-MPC System TSC shell show that the peak temperature is below 156°C [312°F]. Because the peak temperatures for steel subcomponents exposed to sheltered, outdoor air, and embedded environments are below the temperature required to cause reductions in toughness, thermal aging is not considered to be credible for these subcomponents, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.1.1.9 Radiation Embrittlement

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

Neutron irradiation has the potential to increase the tensile and yield strength and decrease the toughness of carbon and alloy steels [3.9.119]. Neutron fluence levels greater than 10^{19} neutrons/square centimeter (n/cm^2) [6.5×10^{19} n/in.^{2}] are required to produce a measurable degradation of the mechanical properties [3.9.119; 3.9.130]. For dry cask storage, a neutron flux of 10^4–10^6 n/cm^2-s [6.5×10^4 – 6.5×10^6 n/in.^{2}-s] is typical [3.9.142]. At these flux levels, the accumulated neutron dose after 60 years is about 10^{13}–10^{15} n/cm^2 [6.5×10^{13}–6.5×10^{15} n/in.^{2}], which is four to six orders of magnitude below the level that would degrade the fracture resistance of carbon and alloy steels. In addition, neutron flux decreases with time during storage, which will limit the radiation effects. Thus, radiation embrittlement of steel exposed to any environment is not a credible aging mechanism.

The low levels of exposure to significant neutron fluence of NAC-MPC System steel components in the VCC and TFR in all environments is not a credible aging mechanism, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.1.1.10 Stress Relaxation

Stress relaxation of bolting or other tightening subcomponents is the steady loss of elastic stress in a loaded part due to atomic movement at elevated temperature. In the NAC-MPC System, steel bolting is only used for the securing of the VCC lid and the TFR retaining ring, and the bolt torques applied and required are very low.

Steel Subcomponents Exposed to Air-Outdoor and Sheltered Environments

NAC-MPC System VCC lid bolting in outdoor environments is not considered to be exposed to sufficiently high temperatures to cause stress relaxation. Similarly, NAC-MPC System TFR bolting in indoor/outdoor environments is not considered to be exposed to high temperatures for an enough time to cause stress relaxation. There are no NAC-MPC System bolts used in sheltered environments. Thus, for steel bolting exposed to outdoor air and indoor/outdoor air environments, aging management is not required during the 40-year period of extended operation.

3.2.1.1.11 Wear

Contact wear results from the repeated mechanical stressing of the surface of a body sliding on another body. For the NAC-MPC System TFR exposed to air-indoor/outdoor, the TFR shield doors experience sliding contact with the TFR door rails during TSC transfer operations. Both SSC subcomponents are constructed of A350 LF2 low alloy steel. Thus, wear of these steel subcomponents is considered to be credible, and therefore, aging management is required during the 40-year period of extended operation. Aging management is addressed in the Transfer Cask AMP as discussed in Section 3.4 to evaluate the effects of the wear of these subcomponents.

3.2.1.2 Stainless Steel

Austenitic and precipitation-hardened stainless steels are used in constructing NAC-MPC System subcomponents. The NAC-MPC System stainless steel components include the TSC shell weldment, structural and shield lids, closure lid (MPC-LACBWR) and fuel basket components; and VCC inlet and outlet screen assemblies, and baffle cover plate. These SSC subcomponents are exposed to air-outdoor, sheltered, encased, and helium environments.

3.2.1.2.1 General Corrosion

Stainless steels exhibit passive behavior in all dry storage environments, resulting in negligible general corrosion rates [3.9.83]. As such, general corrosion of stainless steel exposed to all environments is not considered to be credible, and therefore, aging management is not required for the NAC-MPC System during the 40-year timeframe of the period of extended operation.

3.2.1.2.2 Pitting and Crevice Corrosion

Pitting corrosion is a localized form of corrosion that is confined to a point or small area of a metal surface [3.9.75], and crevice corrosion occurs in a wetted environment when a crevice exists that allows a corrosive environment to develop in a component [3.9.97]. In the NAC-MPC System, crevice corrosion is a potential credible aging effect as the bottom plate of the TSC rests on a stainless-steel sheet, which protects the base of the TSC from potential contamination from the carbon steel pedestal plate and is discussed below. Stainless steels are susceptible to pitting corrosion with chloride being the most common agent for initiation [3.9.83].

Stainless Steel Subcomponents Exposed to Air-Outdoor and Sheltered Environments

The potential to form aqueous electrolytes for subcomponents exposed to outdoor and sheltered environments is present, either via direct exposure to precipitation or by deliquescence of deposited salts. These electrolytes could be conducive to pitting and crevice corrosion of stainless steel. Atmospheric corrosion of stainless steels typically proceeds in the form of localized corrosion [3.9.54; 3.9.141; 3.9.144]. However, experimentally measured penetration rates for pitting and crevice corrosion are quite low. Stainless steel exposed to a saturated NaCl steam mist at 60°C [140°F] and 95 percent relative humidity [3.9.129] yielded maximum penetration rates of 0.02 mm/yr. [8 mils/yr.] for pitting and 0.03 mm/yr. [11 mils/yr.] for crevice corrosion. These maximum rates suggest that penetration of a 15-mm [0.59-in.]-thick canister wall by pitting or crevice corrosion would require 750 years and 495 years, respectively. Davison et al. [3.9.57] reported pitting penetration of 0.028 mm [1.1 mils] after 15 years, which yields a penetration rate of 0.0019 mm/yr. [0.075 mils/yr.]. Based on the penetration rate and using the penetration depth versus time equations from NRC [3.9.4] as follows:

$$d = At^{-n} \text{ and } n = 0.33 \text{ to } 0.5,$$

with n=0.5 yields a penetration time for a 16.5 mm (0.65 in.) thick canister wall of > 20,000 years. Therefore, pitting corrosion is not expected to produce damage to the TSC stainless steel components in a 60-year timeframe. However, pitting corrosion is known to be a precursor to stress corrosion cracking (SCC) as all SCC cracks started at the bottom of pits. In addition, the penetration rate for the sacrificial plate located between the bottom of the TSC and the VCC baffle baseplate is significantly greater than the 60-year timeframe. Therefore, effects of pitting and crevice corrosion over the 40-year period of extended operation of stainless-steel subcomponents exposed to sheltered air is considered to be credible, and aging management is required during the 40-year timeframe of the period of extended operation. The AMP proposed for pitting and crevice corrosion monitoring is contained in the TSC Localized Corrosion and SCC AMP as discussed in Section 3.4.

Stainless Steel Subcomponents Exposed to Helium and Encased Environments

Stainless steel SSC subcomponents exposed to helium are not susceptible to pitting and crevice corrosion due to the lack of halides. Because of limited water and oxygen, stainless steel is also not susceptible to pitting and crevice corrosion in fully encased environments. As such, pitting and crevice corrosion of stainless steel exposed to helium and fully encased environments are not considered to be credible for the NAC-MPC System, and therefore, aging management is not required during the 40-year timeframe of the period of extended operation.

3.2.1.2.3 Galvanic Corrosion

Galvanic corrosion occurs when two dissimilar metals or conductive materials are in physical contact in the presence of a conducting solution [3.9.37; 3.9.84]. Galvanic corrosion is not a credible aging mechanism for stainless steel components in a helium, encased or embedded environment as graphite containing materials or other conductive materials are not used in the fabrication, assembly or operation of the NAC-MPC System TSC and fuel basket components, and there is no conduction solution available after draining, vacuum drying, and backfilling the TSC with high purity helium. Therefore, aging management for galvanic corrosion is not required for NAC-MPC System TSC and fuel basket stainless steel components during the 40-year period of extended operation.

3.2.1.2.4 Microbiologically Influenced Corrosion (MIC)

MIC is caused or promoted by the metabolic activity of microorganisms [3.9.58], and microorganisms can live in many environments, such as water, soil, and air, where aerobic bacteria (e.g., iron-manganese oxidizing bacteria, sulfur/sulfide oxidizing bacteria, methane producers, organic acid-producing bacteria), fungi, and algae can develop.

Stainless Steel Subcomponents Exposed to Helium and Encased Environments

Because of the limited amount of water and nutrients in the helium environments within casks and canisters, and the limited amount of air in the fully encased (steel) environments, MIC of stainless steel is not credible for the NAC-MPC System during the 40-year period of extended operation, and therefore, aging management is not required.

Stainless Steel Subcomponents Exposed to Sheltered and Outdoor Environments

The potential to form aqueous electrolytes for subcomponents exposed to outdoor and sheltered environments is present during the 60-year timeframe, either from direct exposure to precipitation or by deliquescence of deposited salts. These electrolytes could support microbial activity; however, there has not yet been any operating experience of MIC in atmospheric environments where stainless steel surfaces are only intermittently wetted. Due to the absence of any operating experience of MIC damage of stainless steel under atmospheric conditions, MIC is not considered to be credible for the NAC-MPC System, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.1.2.5 Stress-Corrosion Cracking (SCC)

SCC is the cracking of a metal produced by the combined action of corrosion and tensile stress and is highly chemical specific [3.9.93; 3.9.92]. Austenitic stainless steels Type 304 and 304L are susceptible to SCC, under specific environmental conditions, and this susceptibility increases when the material is sensitized [3.9.19]. In the welded condition, the heat-affected zone, which is a thin band located adjacent to the weld, can be sensitized by the precipitation of carbides that extract chromium out of the metal matrix.

The Electric Power Research Institute [3.9.65; 3.9.64] and the Nuclear Decommissioning Authority in the United Kingdom [3.9.128] published review reports on SCC of stainless steel. More recently, the NRC released Information Notice (IN) 2012-20, "Potential for Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage Systems" [3.9.121]. IN 2012-20 describes several incidents in commercial nuclear power plants where SCC of austenitic stainless-steel components was attributed to atmospheric chloride exposure. These events involved components such as emergency core cooling system piping, SNF pool cooling lines, and outdoor tanks. Additionally, IN 2012-20 notes that chlorides may be present in the atmosphere, not only in marine environments but also near cooling towers, salted roads, or other locations. The susceptibility of austenitic stainless steels to SCC tends to increase as the chloride concentration in the solution increases, but the level of chloride required to produce SCC is very low and is dependent on the type of chloride salts present. The material is more resistant to SCC in NaCl solutions but cracks readily in $MgCl_2$ solutions [3.9.83]. Increased temperature and the presence of oxygen tend to aggravate chloride-induced SCC.

Stainless Steel Subcomponents Exposed to Sheltered Environments

The potential to form electrolytes for NAC-MPC System stainless steel subcomponents exposed to sheltered environments is present by deliquescence of deposited salts. These electrolytes could be conducive to SCC of stainless steel. SCC also requires the presence of a tensile stress, which commonly exists at welds originating from fabrication processes.

Stresses well below yield can cause SCC and the required stress for SCC initiation decreases as chloride concentration and temperature increase [3.9.76]. SCC tests were performed with Type 304L C-ring specimens strained to 0.4 or 1.5 percent [3.9.19]. At the strain of 0.4 percent, the stress on the C-ring specimen was approximately equal to the material yield stress. SCC initiation was observed on specimens deposited with 1 or 10 grams/square meters (g/m^2) [0.003 or 0.03 ounces/square foot (oz/ft^2)] of simulated sea salt at both strain levels. Constant load tensile tests were performed on Type 304 between 0.5 and 1.75 times the material yield stress [3.9.110]. Surface chloride concentration was estimated to exceed 10 g/m^2 [0.03 oz/ft^2], while test conditions were 80°C [176°F] at 35 percent relative humidity. Specimens failed at the stress level of 0.5 times the yield stress.

The stainless steel TSC weldment (shell and baseplate) and structural/closure lid are welded as an assembly in the NAC-MPC System. Research [3.9.76] has concluded that the driving stress for SCC of the welded canister is expected to be weld residual stress, considering that the applied stresses are low and residual compressive stresses are believed to be present on the shell outer diameter due to rolling. The referenced calculations indicate that residual stresses parallel to the weld are tensile through-wall and significantly above the original yield strength of the base metal, while those transverse to the weld are either compressive along the outer TSC surface or slightly tensile on the outer diameter but compressive along the midwall. Based on these calculated residual weld stresses, it was concluded that through-wall SCC is most likely to occur transverse to the weld direction. Weld residual stress modeling conducted by the NRC [3.9.120] also indicates that through-wall tensile stresses of sufficient magnitude to support SCC are likely to exist in the weld heat-affected zone.

Because sufficient weld residual stresses and more susceptible material conditions are present near the welds, and aqueous electrolytes conducive to SCC are present in a sheltered environment, the potential for SCC of the welds in the TSC weldment and structural lid is present in the 40-year timeframe of the period of extended operation. Additionally, the SCC initiation times are relatively short [3.9.129] with reported crack growth rates of austenitic stainless steels at the weld heat-affected zones ranging from 0.1 mm/yr. [3.9 mils/yr.] to 0.67 mm/yr. [26.1 mils/yr.]). As a result, through-wall penetration could occur during the 40-year timeframe of the period of extended operation. This is consistent with the observation of outer-diameter-initiated through-wall SCC in stainless steel piping after 20 to 30 years of exposure in marine environments. As such, atmospheric SCC of stainless-steel subcomponents with welds exposed to sheltered air is considered to be credible for the NAC-MPC System, and therefore, aging management is required during the 40-year timeframe of the period of extended operation. The AMP proposed for SCC monitoring is contained in the TSC Localized Corrosion and SCC AMP as discussed in Section 3.4.

Stainless Steel Subcomponents Exposed to Helium and Fully Encased (Steel) Environments

Because of the lack of halides and the small amount of water in helium and fully encased (steel), environments, SCC of stainless steel is not considered to be credible in these environments. Therefore, aging management of stainless-steel subcomponents exposed to helium and fully encased environments is not required during the 40-year timeframe of the period of extended operation.

3.2.1.2.6 Creep

The NAC-MPC System TSC is fabricated from 300 series stainless steels with some basket structural components fabricated from precipitation hardened stainless steels. The impact of creep on the TSC and basket SSCs will focus on the austenitic stainless steels as they have the lowest melting point and minimum creep temperature. Austenitic stainless steels have a melting point of 1,698 K (1,425°C [2,597°F]) and temperatures of at least 679 K (406°C [763°F]) are required to initiate creep in these steel components.

Stainless Steel Subcomponents Exposed to Helium

The highest temperatures within the NAC-MPC System TSC and fuel basket are at locations close to the fuel rods where the environment is helium. The maximum allowable temperature of fuel cladding is limited to 400°C [752°F] at the beginning of storage per ISG-11. This cladding temperature is expected to decrease to around 266°C [510°F] after 20 years and to approximately 127°C [261°F] after 60 years. These estimates depend on many factors, such as the initial heat load of the SNF. Because the fuel rods are the only heat source within the canister, these temperatures provide upper temperature limits for all subcomponents within the TSC. It is apparent from these temperatures that subcomponents within the canister will not reach the 406°C [763°F] minimum temperature that is required for significant creep to occur in austenitic stainless steels.

Similarly, significant creep would also not be expected to occur in the other classes of stainless steel such as the 17-4 PH structural support disks of the basket, which has a higher minimum creep temperature. Hence, creep of TSC stainless steel internals exposed to helium is not credible in the NAC-MPC System, and therefore, aging management is not required during the 40-year period of extended operation.

Stainless Steel Subcomponents Exposed to Sheltered and Fully Encased (Steel) Environments

Because NAC-MPC System stainless steel TSC subcomponents exposed to sheltered and encased environments (e.g., TSC shell weldment, volumes between shield lid and structural lid) experience significantly lower temperatures than those experienced by the internal subcomponents, creep of these stainless-steel subcomponents is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.1.2.7 Fatigue

Spent fuel storage in a NAC-MPC System is a static application and cyclic loading by a purely mechanical means is largely limited to cyclic loads due to thermal effects, such as those caused by daily and seasonal fluctuations in the temperature of the external environment.

The potential for fatigue in the NAC-MPC System TSC and fuel baskets were initially analyzed in the FSAR in accordance with the rules of the ASME Code, Section III, Division 1, Subsection NB and NG, respectively. A TLAA has been prepared as discussed in Section 3.3 to support a determination that fatigue will not challenge ITS functions of the NAC-MPC System TSC SSC subcomponents in the 40-year period of extended operation.

3.2.1.2.8 Thermal Aging

The microstructures of the NAC-MPC System TSC and fuel basket assembly stainless steel components will change, given sufficient time at temperature, and these changes may alter the material's strength and fracture toughness. This process is commonly called thermal aging. For stainless steel subcomponents, the thermal aging process differs for welded and non-welded subcomponents.

Welded Stainless Steel Subcomponents Exposed to Helium

The ferrite present in austenitic stainless-steel welds can transform by spinodal decomposition to form Fe-rich alpha and Cr-rich alpha prime phases, and further aging can produce an intermetallic G-phase. The spinodal decomposition and the formation of the intermetallic G-phase takes place during extended exposure to temperatures between 300 and 400°C [572 and 752°F] [3.9.28; 3.9.50]. The maximum expected temperature of fuel cladding has been estimated to be 400°C [752°F] at the beginning of storage [3.9.94]. This cladding temperature is expected to decrease to around 266°C [510°F] after 20 years and to approximately 127°C [261°F] after 60 years. Based on these temperature estimates, subcomponents located inside the canister and near the fuel could be above the 300°C [572°F] minimum temperature required for these phase changes. Because the phase transformations take place only within the ferrite phase, they increase the hardness and reduce the toughness of the ferrite phase but do not alter the mechanical properties of the austenite phase. Hence, the degree of embrittlement of a weld will depend on many factors, including the amount and distribution of ferrite present in the weld and the time spent within the 300 to 400°C [572 and 752° F] temperature range.

In the NAC-MPC System fuel basket assembly, the only welded components close to the fuel assemblies are the fuel tubes and the fuel tube cladding, and Maine Yankee site-specific damaged fuel cans. NUREG/CR–6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless-Steel Pipe Welds," concluded that thermal aging produced moderate decreases (no more than 25 percent) in the upper shelf Charpy impact energy and relatively small decreases in the fracture toughness of a wide range of austenitic welds. Although the phase changes associated with thermal embrittlement of austenitic stainless-steel welds could take place in subcomponents near the fuel within the 60-year timeframe, the minor reductions in fracture toughness that would be produced in the weld indicate that this is not a credible aging mechanism for subcomponents in proximity to the fuel rods, and therefore, aging management is not required for the 40-year period of extended operation.

In the NAC-MPC System TSC, the other welded components exposed to the helium environment is the TSC shell weldment, shield support ring and shield lid. These components are located at the periphery of the fuel basket and experience temperatures significantly below 300°C, which is the minimum temperature for embrittling phase changes. Due to these lower temperatures, thermal aging will not produce any degradation in these subcomponents, and therefore, aging management is not required during the 40-year timeframe of the period of extended operation.

Nonwelded Stainless Steel Subcomponents Exposed to Helium

Because the phase changes described previously occur only within the ferrite-containing, heat-affected zone of a weld, embrittlement will not occur in nonwelded NAC-MPC System TSC fuel basket austenitic stainless-steel subcomponents. The only significant thermal aging possible in nonwelded austenitic stainless steels would be a decrease in strength due to a decrease in dislocation density, recrystallization, and an increase in grain size. These processes occur during annealing at temperatures above 1,000°C [1,832°F]. For the 17-4 PH stainless steel structural support disks, the maximum long-term storage temperature at full design heat load is 538°F (average temperature is 358°F) per the FSAR [3.9.1.a - 3.9.1.m], which is well below the ASME Code, Section II, Appendix D allowable temperature of 650°F for this material. Thus, thermal aging of nonwelded stainless steel, including 17-4 PH stainless steel structural disks, is not credible, and therefore, aging management is not required during the 40-year period of extended operation.

Welded Stainless Steel Subcomponents Exposed to Sheltered and Encased (Steel) Environments

Because the peak temperatures for NAC-MPC System TSC stainless steel subcomponents exposed to sheltered and fully encased steel environments are below the temperature required for the phase changes associated with thermal embrittlement of austenitic stainless-steel welds, thermal aging is not considered to be credible for these subcomponents, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.1.2.9 Radiation Embrittlement

Embrittlement of metals may occur under exposure to neutron radiation. 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.

The neutron fluence that the NAC-MPC System TSC and fuel basket components are exposed to are five to seven orders of magnitude below the level identified by the NRC [3.9.4] that would degrade the mechanical properties of the TSC stainless steel components. As such, radiation embrittlement of stainless steel exposed to any environments is not credible.

3.2.1.2.10 Stress Relaxation

In the NAC-MPC System, high strength steel bolts are used to secure the VCC lid to the VCC following TSC loading operations in the air-outdoor environment. The loss of initial applied stress in austenitic stainless-steel bolting due to stress relaxation is negligible at temperatures below 300°C [572°F]. The temperature is significantly below these temperatures at the VCC lid bolt locations, and therefore, stress relaxation of the VCC lid stainless steel bolts is not considered to be credible. Therefore, aging management for stress relaxation of the VCC lid bolts is not required during the 40-year period of extended operation.

3.2.1.2.11 Wear

There are no NAC-MPC System stainless steel components that slide against each other during normal loading and storage operations, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.1.3 Aluminum Alloys

In the NAC-MPC System, SB209 6061-T651 aluminum alloy is used in the TSC fuel basket assembly as heat transfer disks, and the heat transfer disks provide an ITS function to transmit the decay heat from the SNF to the TSC shell. The heat transfer disks do not provide a structural ITS function for the basket assembly. These are the only aluminum ITS components included in the NAC-MPC System design.

3.2.1.3.1 General Corrosion

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

Aluminum Subcomponents Exposed to Helium

Following vacuum drying of the NAC-MPC System TSC, there is very little residual water in the cantier internal environment. Assuming a residual water content of 1 liter (L) [0.26 gallon (gal)], Jung et al. [3.9.94] calculated that oxidation of all aluminum in the basket assembly is limited to just 0.54 g [0.019 oz.], which is equivalent to a 20- or 2-μm (0.79 or 0.079-mils] - thick layer of aluminum over a surface area of 100 or 1,000 cm^2 [15.5 or 155 $in.^2$]. In the NAC-MPC System fuel baskets, the total surface area for the 0.5-inch-thick heat transfer disks is > 25,000 in^2. As a result, sufficient general corrosion to challenge the SSC heat transfer ITS functions of the aluminum disks is not credible, and therefore, aging management is not required during the 40-year period of extended operation in a helium environment.

3.2.1.3.2 Pitting and Crevice Corrosion

Pitting corrosion is a localized form of corrosion that is confined to a point or small area of a metal surface and crevice corrosion occurs in a wetted environment when a crevice exists that allows a corrosive environment to develop in a component. Aluminum and its alloys form a passive film on the surface. Localized corrosion in the form of pitting or crevice corrosion could occur for these passive aluminum materials, especially in the presence of halides.

Aluminum Subcomponents Exposed to Helium

Pitting and crevice corrosion of aluminum is not considered to be credible in a helium environment because of the lack of moisture and halides in the helium environment within the NAC-MPC System TSC. Therefore, aging management of pitting and crevice corrosion is not required for aluminum exposed to a helium environment during the 40-year period of extended operation.

3.2.1.3.3 Galvanic Corrosion

Galvanic corrosion occurs when two dissimilar metals or conductive materials are in physical contact in the presence of a conducting solution [3.9.37; 3.9.84]. In NAC-MPC System TSC basket assemblies, galvanic coupling may exist between aluminum and stainless-steel assembly components.

Aluminum Subcomponents Exposed to Helium

There is very little residual water within a NAC-MPC System TSC following drying. Assuming a residual water content of 1 L [0.26 gal], a loss of heat transfer disk material thickness due to material thinning from oxidation is a very small fraction of the aluminum used inside the system. In conclusion, loss of material due to galvanic corrosion in helium environments is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.1.3.4 Microbiologically Influenced Corrosion (MIC)

MIC is corrosion caused or promoted by the metabolic activity of microorganisms [3.9.58]. Microorganisms can live in many environments, such as water, soil, and air, where aerobic bacteria (e.g., iron-manganese oxidizing bacteria, sulfur/sulfide oxidizing bacteria, methane producers, organic acid-producing bacteria), fungi, and algae can develop.

Aluminum Subcomponents Exposed to a Helium Environment

Because of the limited amount of water and nutrients in the helium environment within the NAC-MPC System TSC, MIC of aluminum is not credible for the 40-year period of extended operation, and therefore, aging management is not required.

3.2.1.3.5 Creep

Thermal activation is insufficient to produce significant creep at temperatures below 0.4Tm, where Tm is the melting point of the metal in Kelvin [3.9.46]. With melting points of 911 to 930 K (638 to 657°C [1,180 to 1,215°F]), temperatures of at least 364 to 372 K (91 to 99° C [196 to 210°F]) are required to initiate significant creep in aluminum. These temperatures are consistent with Sindelar et al. [3.9.142], which indicates that creep in aluminum is possible at temperatures greater than 100°C [212°F]. Microstructure also plays a significant role in a metal's resistance to creep. Hence, while this 100°C [212°F] minimum temperature for creep is representative for pure aluminum, creep in precipitation hardened aluminum alloys as used in the NAC-MPC System basket assemblies do not become significant until about 200°C [392°F] [3.9.140]. Additionally, at temperatures near these threshold values, high stresses are required to produce creep. High stresses do not exist in the fuel basket non-structural aluminum heat transfer disks, which provide for heat transfer of fuel decay heat as their primary ITS function.

Aluminum Subcomponents Exposed to Helium

The highest temperatures within the NAC-MPC System TSC are at locations close to the fuel rods where the environment is helium. The maximum allowable temperature of fuel cladding has been established to be 400°C [752°F] at the beginning of storage in accordance with ISG-11 [3.9.10]. This cladding temperature is expected to decrease to below 266°C [510°F] after 20 years and to below 127°C [261°F] after 60 years for TSCs loaded with design basis SNF decay heat load. The maximum long-term storage temperature of the aluminum heat transfer disks at full design heat load is 534°F (average temperature is 346°F) per the FSAR [3.9.1.a - 3.9.1.m]. Because the fuel rods are the only heat source within the TSC, these temperatures provide upper temperature limits for all subcomponents. It is apparent from these temperatures that subcomponents within the TSC could be exposed to temperatures above the minimum creep temperatures for aluminum during at least the first 40 years. Subcomponents such as the NAC-MPC System fuel basket heat transfer disks that do not serve a structural function are not expected to be under loads other than their own weight, and the disks weight is supported by the fuel basket's six or eight tie rods. Due to the minimal applied loads, creep of non-structural heat transfer disks will not produce significant damage to affect their ITS function during the 40-year period of extended operation and therefore, aging management is not required.

3.2.1.3.6 Fatigue

The NAC-MPC System storage operation is a static application. However, the aluminum fuel basket heat transfer disks could experience cyclic loads due to thermal effects, such as those caused by daily and seasonal fluctuations in the temperature of the external environment.

Due to the minimal applied loading conditions on the disks and limited cyclic thermal loads as the decay heat of the fuel continues to reduce over time, fatigue of the non-structural heat transfer disks will not produce significant damage to affect their ITS function during the 40-year period of extended operation, and therefore, aging management is not required.

3.2.1.3.7 Thermal Aging

The microstructures of many aluminum alloys will change, given sufficient time at temperature. This process is commonly called thermal aging. The effect of the thermal aging on mechanical properties will depend on the time at temperature and the microstructure and chemical composition of the aluminum components. In the NAC-MPC System SB209 6061-T651 aluminum alloy is used in the TSC fuel baskets to transfer heat.

Aluminum Subcomponents Exposed to Helium

The 6061-T651 aluminum alloy of the heat transfer disks is a precipitation-hardened alloy. The precipitation treatment is performed between 163° C and 204° C [325°F and 399°F]. The maximum allowable temperature of fuel cladding for the NAC-MPC System is < 400°C [752°F] at the beginning of storage per ISG-11. This cladding temperature is expected to decrease to around 266°C [510°F] after 20 years and to approximately 127°C [261°F] after 60 years. It is apparent from these temperatures that the 6061 aluminum alloys may experience significant overaging at a higher temperature than that for precipitation treatment, leading to loss of strength. This annealing will reduce strength, which could be significant for subcomponents that serve a structural function.

As the NAC-MPC System aluminum disks are not structural components, thermal aging of the non-structural heat transfer disks is not expected to be an issue during the 40-year period of extended operation, and therefore, aging management is not required.

3.2.1.3.8 Radiation Embrittlement

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

Alexander [3.9.28] showed that irradiation at 10^{22} n/cm^2 [6.5×10^{22} n/in.^{2}] simulating reactor conditions affected the mechanical properties of aluminum alloy 6061-T651. However, these radiation levels are five to seven orders of magnitude higher than the fluence after dry storage for 60 years, based on the typical neutron flux of 10^4–10^6 n/cm^2-s [6.5×10^4 – 6.5×10^6 n/in.^{2}-s] during dry storage [3.9.142]. Furthermore, the flux of neutrons within the NAC-MPC System TSC decreases with storage time. The low dose and the decrease of neutron flux with time will limit the radiation effects.

Some results from radiation testing of aluminum-based neutron poisons are reported in the literature [3.9.61]. Gamma, thermal neutron, and fast neutron radiation testing of an aluminum-based laminate composite in water for 9 years and exposed to up to 7×10^{11} rad gamma, 3.6×10^{18} n/cm^2 [2.2×10^{19} n/in.^{2}] fast neutron fluence, and 2.7×10^{19} n/cm^2 [1.7×10^{20} n/in.^{2}] thermal neutron fluence showed no change in ultimate strength and no other signs of physical deterioration except for severe oxidation because of the presence of water. Also, radiation testing of an aluminum-based, sintered composite subjected to up to 1.5 × 10^{20} n/cm^2 [9.7 × 10^{20} n/ $in.^2$] fast neutron fluence and a maximum of 3.8 × 10^{11} rad gamma exposure showed little change in the yield strength and ultimate strength [3.9.61]. Finally, neutron radiation of borated aluminum to fluences of 10^{17} n/cm^2 [6.5 × 10^{17} n/ $in.^2$] showed no dimensional change or radiation damage [3.9.61]. These test conditions are expected to be more severe than those experienced by aluminum alloys in the extended storage application [3.9.61]. Thus, radiation embrittlement of aluminum heat transfer disks exposed to any environments is expected to be insignificant, and therefore, aging management is not required during the 60-year timeframe.

3.2.1.4 Lead

Lead is used as gamma radiation shielding in the NAC-MPC System TFR where the lead is fully encased in steel shells and thus it is not exposed to water or atmospheric contaminants. Lead is well known to be very resistant to corrosion in a variety of environments. Because there are no credible aging mechanisms that could challenge the ability of lead to perform its shielding functions, aging management of this material is not required during the 40-year period of extended operation.

3.2.2 Neutron Shielding Materials

Neutron shielding typically is provided by either borated or non-borated polymeric, or cementitious materials. Hydrogen and oxygen reduce the energy of the neutrons such that the neutrons are more effectively absorbed by the boron. In the NAC-MPC System both polymeric (NS-4-FR) and cementitious (NS-3) materials may be used. The NS-4-FR is provided with 0.61% of B_4C in the shielding mixture

The degradation and possible relocation of shielding materials is mitigated by encasing or reinforcing materials as is the case for the NAC-MPC System. In the NAC-MPC System, the NS-4-FR shielding provided for the TFR is fully encased (poured in place) between the inner and outer steel shells and lead brick layer of the transfer cask body assembly. The NS-4-FR and NS-3 materials utilized in the NAC-MPC System VCC shield plugs are also fully encased in a fully encased steel plate structure.

A set of known aging mechanisms with the potential to affect the performance of shielding materials has been identified from reviews of a range of information as detailed in the MAPS report [3.9.4]. Sources of the information include gap assessments for dry cask storage systems, relevant technical literature, and operating experience from nuclear applications [3.9.20; 3.9.14; 3.9.51; 3.9.85; 3.9.142; 3.9.129; 3.9.8]. These mechanisms, which are induced by thermal and irradiation conditions, include boron depletion, thermal aging, and radiation embrittlement are discussed below.

3.2.2.1 Boron Depletion (Borated Materials)

The boron concentration in the neutron shields decreases as boron atoms in the borated materials absorb neutrons. Boron-10 nuclei capture neutrons, yielding excited boron-11 nuclei, which in turn decay into high-energy alpha particles and lithium-7 nuclei. The neutron shielding material will lose one boron-10 atom per such a reaction. Significant depletion of boron-10 atoms may occur over time if the shielding material is exposed to sufficient neutron fluence.

A TLAA has been prepared to document the neutron shielding performance of the NAC-MPC System due to boron depletion of the NS-4-FR B_4C during the 40-year period of extended operation as described in Section 3.3. It is noted that there is no boron-10 in the NS-3 material used in the NAC-MPC shield plugs, and therefore, there is no depletion effects on this neutron shielding material.

3.2.2.2 Thermal Aging

Polymers may be susceptible to heat-induced changes to material properties and configuration due to several mechanisms. At elevated temperatures, the long chain backbone of a polymer can undergo molecular scission (breaking) and cross linking. Also, gaseous products may be formed, including H_2, CH_4, and CO_2. These reactions may cause embrittlement, shrinkage, decomposition, and changes in physical configuration (e.g., loss of hydrogen or water) [3.9.352; 3.9.164]. Shrinkage and embrittlement can locally displace shielding material and potentially diminish shielding effectiveness, although this may be mitigated in part by reinforcement materials within the polymer matrix and the support provided by the encasing metal. Because many polymers are known to degrade at elevated temperatures, thermal aging for polymer-based neutron-shielding materials is a credible aging mechanism.

Therefore, a TLAA has been prepared as discussed in Section 3.3 to evaluate the performance of the NS-4-FR in the NAC-MPC System TFR and VCC shield plug installations based on maximum temperatures during operations versus historic thermal testing results to show the continued performance of their important to safety shielding functions during the 40-year period of extended operation.

The cementitious BISCO NS-3 shielding material is used in some of the NAC-MPC System VCC shield plugs as an option in place of the NS-4-FR. There is a potential of NS-3 experiencing some loss of hydrogen (neutron moderator) when exposed to elevated temperatures. However, the material is subjected to only moderate temperature during storage operations. The maximum NS-3 temperature for the NAC-MPC System design basis decay heat load of 17.5 kW is 160°F. During the storage period, the temperatures will continue to decrease as the decay heat of the fuel is reduced with time. As a result, thermal aging of the NS-3 shielding material is not considered to be a credible aging mechanism in the VCC shield plug and therefore, aging management is not required during the 40-year period of extended operation.

3.2.2.3 Radiation Embrittlement

Like the thermal aging mechanism discussed above, radiation can alter polymer structures by molecular scission and cross linking to reduce ductility, fracture toughness, and resistance to cracking [3.9.163; 3.9.162]. For example, the threshold for radiation embrittlement has been found to be about 10^6 rad for polyethylene and significantly lower for other polymers, such as polytetrafluoroethylene [3.9.7]. Depending on the dry cask storage system design and the specific

SNF, this dose can be reached in 10–100 years. Embrittlement can locally displace shielding material and potentially reduce shielding effectiveness, although this may be mitigated, in part, by the support provided by the encasing metal as is the case for the NAC-MPC System transfer cask neutron shielding and VCC shield plug neutron shielding. As a result, radiation embrittlement of polymer-based neutron-shielding materials is a credible aging mechanism during the 60-year timeframe.

Therefore, NAC has prepared a TLAA to evaluate the continued ITS performance of the neutron shielding materials of the NAC-MPC System due to radiation embrittlement of the NS-4-FR and NS-3 in the VCC shield plug and the NS-4-FR of the transfer cask during the 40-year period of extended operation as described in Section 3.3.

3.2.3 Neutron Poison Materials

Subcriticality of the SNF in the NAC-MPC System is maintained, in part, by the placement of Boral® neutron absorbers, or poison plates, around the fuel assemblies. The Boral® plates are exposed to a helium environment in the TSC fuel basket, where temperature and radiation levels are high because of their proximity to the fuel assemblies. The TSC helium environment could also include small amounts of residual moisture left after the drying operations.

A list of known aging mechanisms that have the potential to affect the performance of Boral® neutron poison plates was identified from reviews of a range of information sources, including gap assessments for dry storage systems, relevant technical literature, and operating experience from nuclear and nonnuclear applications [3.9.20; 3.9.14; 3.9.51; 3.9.85; 3.9.142; 3.9.129]. These mechanisms, which are induced by various physicochemical, thermal-mechanical, and irradiation conditions, include general corrosion, galvanic corrosion, wet corrosion and blistering, creep, thermal aging, radiation embrittlement, and boron depletion.

The laminate composite of Boral® consist of: (i) a core of uniformly distributed boron carbide and aluminum alloy particles; and (ii) a surface cladding of aluminum alloy on both sides of the core. Of the identified potential aging mechanisms for neutron poison plates listed above, wet corrosion and blistering are the only mechanisms considered to be credible for Boral®, because only this material has porosity that can trap water and initiate this mechanism. Detailed discussions of all aging mechanisms for Boral® are provided below.

3.2.3.1 General Corrosion

Because aluminum is present and used as an outer cladding (Boral®)), the degree of general corrosion is largely governed by the corrosion of aluminum. As discussed in Section 3.2.1.3.1 for NAC-MPC System aluminum heat transfer disks, aluminum forms a protective oxide film at temperatures below approximately 230°C [446°F]. Above this temperature, the protective film no longer forms if water or steam is present. As such, general corrosion of aluminum is possible if aluminum were exposed to moisture in the internal TSC helium environment. However, there is very little residual water in the TSC internal environment following drying. Assuming a residual water content of 1 L (0.26 gal), Jung et al. [3.9.94] calculated that oxidation of all aluminum in the basket assembly is limited to 0.54 g (0.019 oz), which is equivalent to a 2-μm (0.079-mils)-thick layer of aluminum over a surface area of 1,000 cm^2 (155 $in.^2$). Thus, the potential for material thinning from oxidation is a very small fraction of the aluminum Boral® poison materials used inside the NAC-MPC System. As a result, general corrosion is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.3.2 Galvanic Corrosion

Galvanic corrosion occurs when two dissimilar metals or conductive materials are in physical contact in the presence of a conducting solution. The Boral® neutron poison materials used inside the NAC-MPC System TSC fuel basket can be in galvanic contact with stainless steel, where aluminum is less noble.

As discussed above in the evaluation of general corrosion, there is very little residual water within the TSC following drying. Thus, there is a limited potential for the presence of a conducting solution that can support galvanic corrosion. As a result, loss of material due to galvanic corrosion is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.3.3 Wet Corrosion and Blistering

The core of aluminum-boron carbide laminate composites is not fully sintered and, as a result, can have a porosity of 1 to 8 percent with varying degrees of interconnectivity among pores. This may allow water ingress into the core, where the water can react with the aluminum to form aluminum oxide and hydrogen gas [3.9.61; 3.9.156]. Blistering has been observed in the Boral® cladding in wet and dry storage applications. Tests simulating the wetting and vacuum drying cycles during TSC closure operations show that Boral® can form blisters in the aluminum cladding because of water ingress through its exposed edges [3.9.157]. The blisters are characterized by a local area where the aluminum cladding separates from the underlying boron carbide-aluminum core, and the cladding is physically deformed outward.

Although wet corrosion and blistering may occur, this aging mechanism has not been observed to reduce the neutron absorbing capability of Boral® in spent fuel pool surveillance coupons [3.9.61]. It is equally important to note that, because only a trace amount of water will be left in the TSC after vacuum drying and helium backfill, wet corrosion and blistering will be minimal in a dry TSC. Therefore, wet corrosion and blistering are not considered to be an aging mechanism requiring aging management, and therefore, aging management is not required for Boral® in the NAC-MPC System with respect to criticality safety during the 40-year period of extended operation.

3.2.3.4 Boron Depletion

Boron depletion refers to the loss of the capability of a material to absorb neutrons when the neutron fluence significantly consumes boron-10 atoms. Neutron poison plates typically contain 10^{19} to 10^{21} boron-10 atoms/cm^2 [6.5×10^{19} to 10^{21} boron-10 atoms/$in.^2$] [3.9.61]. A neutron flux of 10^4–10^6 n/cm^2-s [$6.5 \times 10^4 - 6.5 \times 10^6$ n/$in.^2$-s] is typical for dry cask storage. Under a neutron flux, boron-10 nuclei capture neutrons, yielding excited Boron-11 nuclei, which, in turn, decay into high-energy alpha particles and lithium-7 nuclei. In this nuclear reaction, one neutron would deplete one boron-10 atom. At typical levels of neutron flux and boron-10 concentration, the neutron dose after 60 years would deplete at most 0.0002 percent of the available boron-10 atoms. Using the highest expected neutron flux and the lowest boron-10 concentration as a worst-case scenario, only 0.02 percent of the available boron-10 atoms would be depleted after 60 years, which is too small to challenge the criticality control function of the neutron poisons. As such, boron depletion for Boral® is not expected to result in significant changes in the criticality control function. As such, boron depletion is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

Although the above generic evaluation does not identify boron depletion as a significant aging mechanism, a TLAA has been prepared to document the criticality safety of the NAC-MPC System due to limited boron depletion of the Boral® during the 40-year period of extended operation as described in Section 3.3.

3.2.3.5 Creep

As discussed previously, significant creep occurs at temperatures above $0.4T_m$, where T_m is the melting point of the metal in Kelvin [3.9.46]. At these temperatures, plastic deformation or distortion can occur over long times, even under stresses that normally would not be considered enough to cause yielding of the material. Because aluminum is present as an external cladding in the neutron poison plates, and aluminum has a lower melting point than the other portions of the material microstructures (e.g., B_4C), the creep behavior of poison materials is governed by the behavior of aluminum. Applying the $0.4T_m$ rule, the critical creep temperature for aluminum is 100°C [212°F].

The highest temperatures within the NAC-MPC System TSC are at locations close to the fuel rods. For example, the maximum allowable temperature of the cladding on the fuel rods in the NAC-MPC System has been calculated to be less than 400°C [752°F] at the beginning of the storage period in accordance with ISG-11. Cladding temperatures are expected to decrease to approximately 266°C [510°F] after 20 years and 127°C [261°F] after 60 years [3.9.94]. These estimates depend on many factors, such as the initial heat load of the SNF. It is apparent from these temperatures that subcomponents within the TSC could be exposed to temperatures above the minimum creep temperatures for aluminum during at least the first 40 years.

Because temperatures within the NAC-MPC System TSC have the potential to exceed the minimum creep temperature of aluminum, it is necessary to consider the load applied to the subcomponent to determine whether significant creep deformation will occur, as well as the specific application to determine whether the creep affects safety. The NAC-MPC System fuel basket Boral® neutron poison plates do not serve a structural function and only support their own weight. In addition, the weight of the Boral® plates are also supported by the stainless-steel fuel tubes and stainless-steel sheathing. Due to the minimal applied loads and presence of adjacent supporting structures, the impact of creep on the criticality control function of the Boral® neutron poison plates in the NAC-MPC System is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.3.6 Thermal Aging

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

Also, as discussed above for the creep mechanism, decreases in strength due to thermal aging are not expected to affect the criticality control function of the poison plates, because they typically do not serve a structural function and may be supported by adjacent structures. Consequently, thermal aging of NAC-MPC System neutron poison materials is not considered to be credible, and therefore, aging management is not required over the 40-year period of extended operation.

3.2.3.7 Radiation Embrittlement

As discussed previously, embrittlement of metals may occur under exposure to radiation. Neutron radiation (rather than gamma radiation) has the greatest potential to cause this phenomenon. Depending on the neutron fluence, radiation can cause changes in mechanical properties such as loss of ductility, fracture toughness, and resistance to cracking. Research has shown that pure aluminum had increased strength but decreased ductility after being irradiated to fast neutron fluences (energy greater than 0.1 MeV) in the range of 1 to 3 × 10^{22} n/cm^2 [6.5 to 19.4 × 10^{22} n/in.^{2}] from a research reactor for 8 years [3.9.68]. However, these radiation levels are five to seven orders of magnitude higher than the fluence after dry storage for 60 years, based on the typical neutron flux of 10^4–10^6 n/cm^2-s [6.5 × 10^4 – 6.5 × 10^6 n/in.^{2}-s] in a spent fuel dry storage cask [3.9.142].

Gamma, thermal neutron, and fast neutron radiation testing of Boral® in water was performed for 9 years [3.9.61]. With exposures of to up to 7 × 10^{11} rad of gamma, 3.6 × 10^{18} n/cm^2 [2.3 × 10^{19} n/in.^{2}] fast neutron fluence, and 2.7 × 10^{19} n/cm^2 [1.7 × 10^{20} n/in.^{2}] thermal neutron fluence, the specimen showed no change in ultimate strength and no other signs of physical deterioration, except for severe oxidation because of the presence of water. Also, radiation testing of a sintered composite subjected to up to 1.5 × 10^{20} n/cm^2 [9.7 × 10^{20} n/in.^{2}] fast neutron fluence and a maximum of 3.8 × 10^{11} rad gamma exposure showed little change in the yield strength and ultimate strength. These test conditions are more severe than those experienced by Boral® neutron poison in the extended NAC-MPC System application. Therefore, radiation embrittlement of and Boral® is not considered to be credible. Consequently, aging management of Boral® neutron poison in the MPC TSC fuel baskets is not required during the 40-year period of extended operation.

3.2.4 Concrete Overpacks

The concrete overpacks for the stored canister in the NAC-MPC System are identified as Vertical Concrete Casks (VCCs) and the VCCs include various structural subcomponents constructed of concrete and reinforcing steel. These subcomponents may be exposed to several environments, such as outdoor air or they may be sheltered or embedded in concrete. The environment also includes elevated temperatures due to heat released by the SNF and radiation, with dose rates depending on the SNF characteristics (e.g., burnup and age of fuel), exposure time, and location of the subcomponent. Potential aging mechanisms for the VCC subcomponents were identified from reviews of gap assessments of dry storage systems, relevant technical literature, American Concrete Institute (ACI) guides and reports, and operating experience from nuclear and nonnuclear applications. Additional mechanisms were identified during an NRC concrete expert panel workshop [3.9.232]. Thermal, mechanical, chemical, and irradiation-induced degradation mechanisms were identified as follows:

- freeze and thaw
- creep
- reaction with aggregates
- aggressive chemical attack
- corrosion of reinforcing steel (also addressed in Section 3.2.1.1)
- shrinkage
- leaching of calcium hydroxide
- radiation damage
- fatigue
- dehydration at high temperature
- microbiological degradation
- delayed ettringite formation
- salt scaling

Potential mechanisms were refined by considering the thermal, mechanical, chemical, and irradiation conditions specific to each subcomponent. This process eliminated several mechanisms from consideration for some subcomponents in NAC-MPC System VCC AMR Tables 3.2-4, 3.2-5, and 3.2-6. Structural steel subcomponents were also evaluated as documented in Section 3.2.1.1. Potential aging mechanisms for each subcomponent material and the technical bases for those requiring aging management are included in the following sections.

3.2.4.1 Concrete

3.2.4.1.1 Freeze and Thaw

Concretes Exposed to Outdoor Environments Above the Freeze Line

Concretes that are nearly saturated with water can be damaged by repeated freezing and thawing cycles in environments with weathering indexes (i.e., the product of the average annual number of freezing cycle days and the average annual winter rainfall in inches) on the order of 100 day-in./yr. or greater. For environments with weathering indexes less than 100 day-in./yr. freeze and thaw degradation is not significant. Freeze and thaw damage has been observed in outdoor concrete structures in nuclear power plants [3.9.13; 3.9.21]. Because water expands when freezing, fully or mostly saturated concrete will experience internal stresses from the expanding ice, which can cause concrete cracking or scaling when pressures exceed the concrete tensile strength [3.9.171; 3.9.243; 3.9.221; 3.9.248; 3.9.200]. The degradation mode would initiate at the outer concrete surface of the concrete cask system exposed to outdoor environments, primarily at horizontal surfaces where water ponding can occur.

Operating experience has identified freeze and thaw damage in the roofs of NUHOMS concrete storage modules at the Three Mile Island Unit 2 (TMI-2) and the Millstone independent spent fuel storage installation (ISFSI) [3.9.21]. It is expected that freeze and thaw cycle damage would be observed. Therefore, freeze and thaw damage is considered credible in concrete exposed to outdoor environments above the freeze line, and aging management is required during the 40-year period of extended operation. The applicable AMP proposed for the potential impacts of freeze/thaw is the Reinforced Vertical Concrete Cask (VCC) Structures AMP as discussed in Section 3.4.

Concretes Exposed to Sheltered Environments Under the Freeze Line

Freeze and thaw degradation of concrete exposed to sheltered environments with low water availability is not considered credible. The NAC-MPC System does not have exposed concrete in a sheltered environment, and therefore, aging management of concrete of the NAC-MPC System in a sheltered environment for freeze and thaw degradation is not required.

3.2.4.1.2 Creep

Creep in concrete is the time-dependent deformation resulting from sustained load [3.9.267]. Cement paste in concrete exhibits creep due to its porous structure and a large internal surface area that is sensitive to water movements. Creep manifests as cracking on the concrete outer surfaces and causes redistributions of internal forces. Factors affecting creep are concrete constituents (composition and fineness of the cement; admixtures; and size, grading, and mineral content of aggregates), water content and water-cement ratio, curing temperature, relative humidity, concrete age at loading, duration and magnitude of loading, surface-volume ratio, and slump [3.9.267; 3.9.231]. However, the most important parameter controlling creep is concrete sustained loading. Creep increases with increasing load and temperature [3.9.222]. However, the creep rate decreases exponentially with time [3.9.192; 3.9.20; 3.9.267]. In summary, in the case of a given concrete mix design, concrete creep is generally understood to be a phenomenon that would affect concrete structures early in the service life under sustained loading. Thus, the age of concrete and the magnitude and duration of sustained loading are the primary factors that determine the magnitude of the creep of concrete [3.9.231]. For example, if a sustained load is applied on 2-year-old and 40-year-old concrete, the 2-year-old concrete will have significantly more creep. Also, the creep in concrete could largely be mitigated by proper design practices, in accordance with ACI 318-05 [3.9.173] or ACI 349-06 [3.9.172]. Furthermore, creep-induced concrete cracks are not generally large enough to reduce the compressive strength of concrete, cause deterioration of concrete, or cause exposure of reinforcing steel to the environment. In a NAC-MPC System, the initial sustained load is low, and no significant change of load is expected during the 40-year timeframe beyond initial licensing. Thus, creep is not considered credible for any environment, and aging management is not required for the NAC-MPC System during the 40-year period of extended operation.

3.2.4.1.3 Reaction with Aggregates

The two most common alkali-aggregate reactions are alkali-silica reaction (ASR) and alkali-carbonate reaction, with ASR being the most common and damaging. ASR is a chemical reaction between hydroxyl ions (present in the alkaline cement pore solution) and reactive forms of silica present in some aggregates (e.g., opal, chert, chalcedony, tridymide, cristabolite, strained quartz). An aggregate that presents a large surface area for reaction (i.e., amorphous, glassy) is susceptible to ASR [3.9.245]. The resulting chemical reaction produces an alkali-silica gel that swells with the absorption of moisture, exerting expansive pressures within the concrete [3.9.202]. ASR damage in the concrete manifests as a characteristic map cracking on the concrete surface [3.9.168]. The internal damage results in the degradation of concrete mechanical properties, and in severe cases, the expansion can result in undesirable dimensional changes. In reinforced concrete, cracks tend to align parallel to the direction of maximum restraint and rarely progress below the level of the reinforcement. In general, ASR is a slow degradation mechanism that can cause serviceability issues and may exacerbate other deterioration mechanisms.

The requisite conditions for initiation and propagation of ASR include (i) a sufficiently high alkali content of the cement (or alkali from other sources, such as deicing salts, seawater, and groundwater), (ii) a reactive aggregate, and (iii) available moisture, generally accepted to be relative humidity greater than 80 percent [3.9.239; 3.9.255]. Studies have shown that ASR increases proportionally to the cement content, alkali content greater than 0.6 percent can accelerate ASR, high calcium oxide content can promote ASR, and the use of various types of admixtures in certain doses can mitigate ASR [3.9.168; 3.9.189]. At higher concentrations of alkali hydroxides, even the more stable forms of silica are susceptible to ASR attack [3.9.271]. Repeated cycles of wetting and drying can accelerate ASR [3.9.174]. As a result, it is desirable to minimize both available moisture and wet-dry cycles by providing good drainage. Moreover, concretes exposed to warm environments are more susceptible to ASR than those exposed to colder environments [3.9.240].

As mentioned earlier, ASR is generally a slow degradation mechanism. ASR may take from 3 to more than 25 years to develop in concrete structures, depending on the nature (reactivity level) of the aggregates, the moisture and temperature conditions to which the structures are exposed, and the concrete alkali content [3.9.258]. The delay in exhibiting deterioration indicates that there may be less reactive forms of silica that can eventually cause deterioration [3.9.225]. Recent operating experience has revealed degradation of the concrete in the Seabrook reactor containment as a result of ASR [3.9.142]. The concrete used at the Seabrook plant passed all industry standard ASR screening tests [3.9.184; 3.9.182] at the time of construction. However, ASR-induced degradation was identified in August 2010. In addition, ASR screening tests are not conducted on each aggregate source but rather in select batches, which increases the risk for use of aggregates of different reactivities when procured from different sources. Due to the uncertainties in screening tests that can effectively be used to eliminate the potential for ASR and previous ASR operating experience at a nuclear facility, the aging mechanism is considered credible in concrete exposed to any environment with available moisture, and therefore, aging management of the NAC-MPC System is required during the 40-year period of extended operation. The applicable AMP proposed for the potential impacts of reactions to aggregates is the Reinforced VCC Structures AMP as discussed in Section 3.4.

3.2.4.1.4 Aggressive Chemical Attack

The intrusion of aggressive ions or acids into the pore network of the concrete can cause various degradation phenomena. The aggressive chemical attack typically originates from an external source of sulfate or magnesium ions as well as acidic environmental conditions. Depending on the type of aggressive chemical, the degradation of concrete can manifest in the form of cracking, loss of strength, concrete spalling and scaling, and reduction in concrete pH.

Concretes Exposed to Outdoor Environments

1) External Sulfate Attack

External sulfate attack is a process whereby ions in species such as K_2SO_4, Na_2SO_4, $CaSO_4$, and $MgSO_4$, which are present in groundwater, seawater, and rainwater, penetrate the concrete and chemically react with alkali and calcium ions to form a precipitate of calcium sulfate in addition to other forms of calcium and sulfate-based compounds (e.g., ettringite). The manifestation of sulfate attack is cracking, increase in concrete porosity and permeability, loss of strength, and surface scaling generated by the expansion associated with the formation of ettringite within the concrete and the pressure generated by the precipitated calcium and sulfate-base compounds inside the concrete pore network [3.9.244; 3.9.129]. Unlike the alkali sulfates, no decalcification of the calcium silicate hydrate phase occurs in the $CaSO_4$ attack. On the other hand, the $MgSO_4$ attack is significantly faster and more thorough than the attack by the other sulfate compounds because of the limited solubility of $Mg(OH)_2$ in the high pH of concrete [3.9.197]. In addition, magnesium ions present in deicing salts can react with calcium silicate hydrate, gradually converting it to magnesium silicate hydrate, which is not cementitious in nature.

Cases of sulfate attack in the field are fairly uncommon, mainly because most transportation regulatory agencies have adopted specifications aimed at preventing this damage mode [3.9.270; 3.9.264]. In particular, degradation due to external sulfate attack has not been reported in nuclear applications. Atkinson and Hearne [3.9.186] developed a concrete service life model to assess degradation due to sulfate attack. Using aggressive soil and groundwater conditions [sulfate concentration of 1,500 ppm as specified in ASME Code Section XI, Subsection IWL [3.9.180] and typical concrete properties (i.e., elastic modulus, roughness factor, Poisson's ratio, and concrete porosity), the model predicts that sulfate damage can occur within 60 years of exposure [3.9.189].

2) Magnesium Attack

Magnesium ions can rapidly replace calcium ions in the silica hydrate compounds. In groundwater, magnesium ions are commonly found in the form of $MgSO_4$. The magnesium ion attack is more commonly observed in arid western U.S. areas and in below-grade structures. At present, there is no stipulation on the threshold concentration of magnesium ions needed to promote damage to concrete structures for nuclear and nonnuclear applications. Because magnesium attack could be part of the sulfate attack, the timeframe implications and exposure conditions are expected to be comparable to those of sulfate attack.

3) Acid Attack

Acids with a pH less than 3 can dissolve both hydrated and unhydrated cement compounds (e.g., calcium hydroxide, calcium silicate hydrates, and calcium aluminate hydrates) as well as calcareous aggregate in concrete without any significant expansion reaction [3.9.210; 3.9.223]. In most cases, the chemical reaction forms water-soluble calcium compounds, which are then leached away by aqueous solutions. The dissolution of concrete commences at the surface and propagates inward as the concrete degrades. The signs of acidic attack are loss of alkalinity (also disturbing of electrochemical passive conditions for the embedded steel reinforcement), loss of material (i.e., concrete cover), and loss of strength.

The extent and rate of concrete degradation depends on the type, concentration and pH of the acidic solution, concrete permeability, calcium content in the cement, the water-to-cement ratio, and the type of cement and mineral admixtures [3.9.238]. Sulfuric acid is particularly aggressive to concrete, because the calcium sulfate formed from the acid reaction will also deteriorate concrete via sulfate attack [3.9.237]. Even slightly acidic solutions that are lime deficient can attack concrete by dissolving calcium from the paste, leaving behind a deteriorated paste consisting primarily of silica gel.

Acids can come from groundwater as well as from acid rain containing SO_2, NO_X, and HCl from polluted regions, which can compromise the durability of concrete [3.9.268]. Acid rain deterioration is dependent on the amount of acid absorption into the concrete, type of acid, mix proportion, and contact time or interval of rainfalls. As such, this degradation mode is expected to affect the concrete shortly after the concrete surface is in contact with the acid solution.

4) Conclusions

In summary, aggressive chemical attack of concretes exposed to outdoor environments is considered to be credible, and therefore, aging management of the NAC-MPC System is required during the 40-year period of extended operation. The applicable AMP proposed for the observation of potential impacts of aggressive chemical attack is the Reinforced VCC Structures AMP as discussed in Section 3.4.

Concretes Exposed to Sheltered Environments

With regard to concrete in sheltered environments, external sources of sulfate, magnesium, and acid entering concrete are considered to be insignificant. In addition, the heat load from the fuel in the NAC-MPC System is expected to aid in drying the interior concrete surfaces, thus decreasing water availability at the concrete surface, which is necessary to promote this degradation mode. Thus, aggressive chemical attack of sheltered concrete of the NAC-MPC System is not considered credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.4.1.5 Corrosion of Reinforcing Steel and Steel Embedments

Concretes Exposed to Outdoor Environments

Corrosion of the reinforcing steel and other steel components embedded in the concrete is mainly caused by the presence of chloride ions in the concrete pore solution and carbonation of the concrete. Chloride attack of concrete structures is well established [3.9.194]. The highly alkaline environment provided by the concrete (normally with pore water pH >13.0) results in the formation of a metal-adherent oxide film on the reinforcement steel bar surface, which passivates the steel [3.9.236]. However, chloride ions may penetrate the concrete matrix and break down the steel passive layer, once the chloride concentration at the reinforcing steel surface exceeds a threshold value, triggering corrosion of the reinforcing steel and shortening the service life of a concrete structure. For instance, chlorides may already exist at low levels within the base mix constituents. In most practical situations, chloride ions penetrate from the outside environment, such as when using deicing salts, from groundwater, and in marine environments. The presence of corrosion products at the steel surface can generate internal stresses within the concrete matrix, causing cracks and spalling of the concrete cover with consequent structural damage.

The threshold chloride concentration in concrete required to promote corrosion of the reinforcing steel depends on the pH of the concrete pore solution. The onset of corrosion can be enhanced when acid attack or concrete carbonation reduces the concrete pH at the steel surface. Thus, the chloride-to-hydroxide ratio is an important parameter in evaluating the steel corrosion. The present literature does not provide a clear agreement on the value of the critical chloride ion concentration required for corrosion initiation.

Concrete durability is directly related to the quality of the concrete, the external concentration of chlorides on the concrete surface, and the reinforcement material. The service life of concretes exposed to chloride attack depends on the concrete cover, the surface chloride concentration, the chloride diffusion coefficient, the type of cementitious material, and the reinforcing steel material. Several service life models have been proposed to determine the durability of concrete subject to chloride-induced corrosion [3.9.249; 3.9.198; 3.9.189].

No cases of corrosion-induced damage of reinforcing steel and steel embedments such as Nelson studs have been reported within the 60-year timeframe for concretes of moderate to high quality such as that achieved in the construction of NAC-MPC VCCs. The corrosion of reinforcing steel and other steel components embedded or partially embedded in concrete exposed to outdoor environments was evaluated in TLAA 30013-2003. The analysis concluded that steel components fully or partially embedded in concrete will maintain their identified safety factors and important to safety functions for the 40-year period of extended operation. The TLAA evaluating steel components fully or partially embedded in VCC concrete structures for the potential impacts of general corrosion is discussed in Section 3.3.

Concretes Exposed to Sheltered Environments

Chloride ingress is expected to be insignificant for steel reinforcement embedded in concrete in sheltered environments with limited exposure to water. In addition, the heat load from the fuel in the NAC-MPC System is expected to aid in drying the interior concrete surfaces, thus decreasing water availability at the concrete surface, which is necessary to promote this degradation mode. Thus, corrosion of reinforcing steel is not considered credible for concrete in a sheltered environment, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.4.1.6 Shrinkage

Shrinkage occurs when hardened concrete dries from a saturated condition to a state of equilibrium in about 50 percent relative humidity [3.9.21]. As excess concrete water evaporates, tensile stresses are induced in the concrete due to internal pressure from the capillary action of water movement, which results in cracking. The factors affecting shrinkage are cement content, water-to-cement ratio, degree of hydration, elastic modulus of aggregates, amount and characteristics of concrete admixtures, temperature and humidity during curing, and size and shape of concrete [3.9.20; 3.9.192; 3.9.225].

According to ACI 209R-92 [3.9.169], over 90 percent of the shrinkage occurs during the first year, reaching 98 percent by the end of the first 5 years. Thus, shrinkage as an effect of aging in exposed concrete is not expected to influence concrete performance after the initial storage or licensing period, because most of the shrinkage will take place early on in the life of the concrete. As a result, shrinkage of concretes exposed to outdoor environments is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.4.1.7 Leaching of Calcium Hydroxide

Concretes Exposed to Outdoor and Sheltered Environments

A constant or intermittent flux of water through a concrete surface can result in the removal or leaching of calcium hydroxide [3.9.85]. Calcium hydroxide leaching is observed in the form of white leachate deposits (calcium carbonate) on the concrete surface. Calcium hydroxide leaching causes loss of concrete strength, converting the cement into gels that have no strength. Leaching also increases the concrete porosity and permeability, making it more susceptible to other forms of aggressive attack. In addition, leaching of calcium hydroxide in concrete lowers the concrete pH, affecting the integrity of the protective oxide film of the reinforcing steel [3.9.63].

The extent of the leaching depends on the environmental salt content and temperature [3.9.14], and it can take place above and below ground. However, the leaching rate is generally slow and controlled by diffusion [3.9.199]. For example, interior inspections conducted at the Calvert Cliffs ISFSI revealed the presence of white-colored stalactite debris in the gap between the heat shield and the concrete ceiling of two sheltered NUHOMS concrete structures after 15–20 years in service. Stalactites are formed when water leaches calcium hydroxide out of the concrete, which precipitates as calcium carbonate on contact with carbon dioxide in the air. The licensee concluded that water entering the outlet vent stack promoted calcium hydroxide leaching [3.9.205]. Other exterior inspections conducted at the Three Mile Island (TMI)-2 ISFSI revealed efflorescence growth on multiple NUHOMS concrete structures exposed to an outdoor environment. The licensee concluded that the efflorescence deposits were formed by water entering freeze and thaw cracks in the anchor blockout holes on the roof of the HSMs. The licensee conducted core sample testing to verify concrete compressive strength. Therefore, operating experience indicates that leaching of calcium hydroxide is a mechanism that can be exacerbated by other degradation mechanisms or designs that do not adequately prevent ingress of precipitation into the sheltered structure. Although the NAC-MPC System does not have similar design or operating features of the NUHOMS, leaching of calcium hydroxide in NAC-MPC System VCC concrete exposed to outdoor and sheltered environments is considered to be credible, and therefore, aging management is required during the 40-year period of extended operation. The applicable AMP proposed for the potential impacts of leaching of calcium hydroxide is the Reinforced VCC Structures AMP as discussed in Section 3.4.

3.2.4.1.8 Radiation Damage

Radiation effects on concrete properties will depend on the gamma and neutron radiation doses, temperature, and exposure period. Gamma radiation can decompose and evaporate water in concrete [3.9.191] and because most of the water is contained in the cement paste, the effect of gamma radiation on cement paste is more significant than on the aggregates. Gamma radiation can also decompose the SiO bond within calcium silicate hydrate. Neutron radiation deteriorates concrete by reducing stiffness, forming cracks by swelling, and changing the microstructure of the aggregates. This consequently reduces concrete strength. The changes in aggregate microstructure also can lead to higher reactivity of aggregates to certain aggressive chemicals.

NUREG/CR-7171, “A Review of the Effects of Radiation on Microstructure and Properties of Concretes Used in Nuclear Power Plants,” provides a comprehensive review of the effects of gamma and neutron radiation on the microstructure and properties of concrete used in nuclear power plants [3.9.154]. Concrete structures have been regarded as being sound as long as the cumulative radiation does not exceed critical levels over the life of the structure. In general, the critical radiation levels to reduce concrete strength and elastic modulus are considered to be approximately 1×10^{19} n/cm^2 [6.5×10^{19} n/in.^{2}] for fast neutrons (neutron energy >1 MeV) and 1-2 × 10^{10} rad [1-2 × 10^8 grays] for gamma rays [3.9.212; 3.9.199; 3.9.215; 3.9.179].

In dry storage system, a neutron flux of 10^4-10^6 n/ cm^2-s [6.5×10^4 – 6.5×10^6 n/in.^{2}-s] is typical [3.9.142]. At these flux levels, the accumulated neutron dose after 60 years is about 10^{13} – 10^{15} n/ cm^2, which is four to six orders of magnitude below the level that would lead to a reduction of concrete strength and elastic modulus. The gamma dose is also expected to be several orders of magnitude less than the limits defined in the above references for the NAC-MPC System design bases. Therefore, aging management of concrete exposed to outdoor and sheltered environments is not considered to be credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.4.1.9 Fatigue

Concrete fatigue strength is defined as the maximum stress that the concrete can sustain without failure under a given number of stress cycles [3.9.20]. Because dry storage is a static application, mechanical cyclic loading is not expected. However, restraint of the concrete from expanding and contracting as it is exposed to rapid changes in temperature will lead to internal stresses in the structure. If the changes in temperature are severe and the resulting strains are sufficient, local plastic deformation can occur. Repeated application of this thermal loading can lead to crack initiation and propagation in low-cycle fatigue.

Concrete fatigue in the dry storage system reinforced concrete may be caused by diurnal and seasonal temperature gradients through the wall of the dry storage system assembly. The inside surface of the concrete wall is hotter than the outside surface of the concrete wall, which causes compressive stresses in the dry storage system concrete near the inside of the concrete wall and tensile stresses in the rebar near the outside of the concrete wall.

Extreme seasonal temperature variations are expected to be significantly higher than diurnal variations, and these can produce higher cyclic stress amplitudes. Assuming ambient temperatures of -40°C [-40°F] (winter) and 52°C [125°F] (summer), the maximum thermal gradient across the dry storage system concrete is expected to be less than 16°C [60°F]. The number of extreme seasonal temperature cycles, conservatively postulated to occur 10 times per year, is 600 over 60 years.

Diurnal temperature fluctuations in ambient air temperatures are assumed to occur once per day. For conservatism, it is assumed that the diurnal temperature fluctuations are 25°C (the largest mean daily change of temperature in the United States). Therefore, the total number of thermal cycles due to diurnal temperature variations in ambient temperatures over 60 years is 21,900 thermal cycles. Thus, the total number of thermal cycles due to seasonal and daily variations over 60 years is 22,500 cycles.

Due to the low level of stresses imposed on the NAC-MPC System VCC, aging management for fatigue of the concrete structure in sheltered or outdoor environments is not considered credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.4.1.10 Dehydration at High Temperature

Exposure of concrete to elevated temperatures can affect its mechanical and physical properties [3.9.242]. It is well known that concretes can degrade at high temperatures due to dehydration of the hydrated cement paste, thermal incompatibility between the cement and aggregates, and physicochemical deterioration of the aggregates [3.9.233]. As the temperature increases to about 105°C [221°F], all evaporable water is removed from the concrete. At temperatures above 105°C [221°F], the strongly absorbed and chemically combined water are gradually lost, with the dehydration essentially complete at 850°C [1,562°F] [3.9.211]. High-temperature degradation in concrete manifests as a change in compressive strength and stiffness, as well as an increase in concrete shrinkage and transient creep, resulting in the formation of cracks [3.9.227; 3.9.232; 3.9.250]. The effect of the elevated temperature is most significant on the concrete's modulus of elasticity, which can decrease up to 40 percent [3.9.203]. Concretes in the temperature range of 20 to 200°C [68 to 392°F] show small changes in compressive strength. Beyond 350°C [662°F], concrete compressive strength decreases rapidly [3.9.233].

In accordance with NUREG-1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" [3.9.122], the NAC-MPC System under maximum decay heat load of 17.5 kW, and maximum ambient temperature and solar load conditions, local concrete temperatures are maintained below 93°C [200°F], and peak temperatures are less than 149°C [300°F]. The effects of thermal dehydration were addressed during the initial NAC-MPC System CoC approval. Because the fuel temperature decreases over time, the design temperature considerations in NUREG-1536 are expected to continue to be adequate.

Thus, dehydration of concrete at high temperature is not considered to be credible for the NAC-MPC System VCC in an outdoor environment, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.4.1.11 Microbiological Degradation

Concretes Exposed to Air-Outdoor and Sheltered Environments

The air-outdoor and sheltered environments may provide favorable conditions for microbiological degradation mechanisms because of the potential presence of moisture. However, the conditions may be intermittent, and there is no evidence that actual concrete subcomponents in the NAC-MPC System environment microbiologically degrade. Thus, microbiological degradation of concretes exposed to outdoor and sheltered environments is not considered credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.4.1.12 Delayed Ettringite Formation

At the initial stage of fresh concrete curing, ettringite, commonly referred to as “naturally occurring ettringite,” is formed by the reaction of tricalcium aluminate and gypsum in the presence of water. The formation of naturally occurring ettringite in fresh concrete is not detrimental to the overall concrete performance. At the still-early stage of concrete curing, the naturally occurring ettringite may convert to monosulfoaluminate if curing temperatures are greater than about 70°C [158°F] [3.9.204]. After concrete hardens, if the temperature decreases below this value, the monosulfoaluminate becomes unstable and, in the presence of sulfates released by the C-S-H gel, ettringite will reform. This mechanism is called “delayed ettringite formation” (DEF), which results in volume expansion and increased internal pressures in the concrete [3.9.204]. Because the concrete has hardened at this stage, the volume expansion leads to cracking and spalling, with greatest severity commonly observed in below-ground structures with elevated temperatures from curing and heat of hydration. DEF has been reported in precast concrete railroad ties in Sweden, cast-in-place concrete structures in the southern United States after 10 years in service, and mass concretes with high cement contents in the United Kingdom. However, to date, no operating experiences exist of DEF degradation for concrete structures at nuclear power plants.

The conditions necessary for the occurrence of DEF are excessive temperatures during concrete placement and curing, the presence of internal sulfates, and a moist environment. ACI 318-05 [3.9.173] indicates that inspection reports shall document concrete temperature and protection during placement when the ambient temperature is above 35°C [95°F]. Protection measures during concrete placement include lowering the temperature of the batch water, cement, and aggregates as referenced in ACI 305R-10 [3.9.167]. As such, following the ACI 318-05, ACI 305R-10, and ACI 308R-01 [3.9.171] guidelines during concrete placement and curing can effectively limit the concrete temperature to below 70°C [158°F], therefore preventing the development of DEF.

NUREG-1536 [3.9.122] cites ACI 349 [3.9.172] and ACI 318 [3.9.173] as applicable codes for the design and construction of the concrete dry storage systems, and were the applicable codes used for the design and construction of NAC-MPC System VCCs. In addition to the adequate placement and curing standards, no occurrences of DEF-related degradation of concrete have been reported in nuclear applications. Thus, DEF of concrete is not considered credible for NAC-MPC System VCCs in outdoor and sheltered environments, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.4.1.13 Salt Scaling

Concretes Exposed to Air-Outdoor Environments Above the Freeze Line

Salt scaling is defined as superficial damage caused by freezing a saline solution on the surface of a concrete body. The damage is progressive and consists of the removal of small chips or flakes of material. Like freeze and thaw damage, salt scaling takes place when concrete is exposed to freezing temperatures, moisture, and dissolved salts. The degradation is maximized at a moderate concentration of salt (e.g., from deicing salts), called the pessimum concentration which is independent of the types of salt species and is about 3 to 4 percent of the solute by weight. The most common deicing salts are sodium chloride and calcium chloride. Other deicing chemicals include magnesium chloride, urea, potassium chloride, ammonium sulfate, and ammonium nitrate.

Salt scaling of concrete roadways, pavements, sidewalks, driveways, decks, and other slabs is a common problem in locations exposed to cyclic freezing and thawing and deicing salts. For vertical surfaces, this damage mechanism is not expected to be operative unless the dry storage system concrete structure is surrounded by standing water containing salts. Therefore, this degradation mode is only expected to initiate and manifest in horizontal structures exposed to outdoor environments where water ponding can occur. The NAC-MPC System does have areas of horizontal structures at the top of the VCC where water ponding can occur. Because salt scaling is closely related to freeze and thaw damage, the timeframe associated with the initiation of salt scaling of concrete could be relevant for both short- and long-term exposures. Therefore, salt scaling damage is considered credible for NAC-MPC System VCC systems within the 60-year timeframe for concrete structures exposed to outdoor air environments above the freeze line, and therefore, aging management is required during the 40-year period of extended operation. The applicable AMP proposed for the observation of potential impacts of salt scaling is the Reinforced VCC Structures AMP as discussed in Section 3.4.

3.2.5 Spent Fuel Assemblies

The spent nuclear fuel (SNF) assembly components evaluated in this section include the zirconium-based and stainless-steel cladding and fuel assembly hardware, which provide structural support to ensure that the spent fuel is maintained in a known geometric configuration. The safety analyses for NAC-MPC System relies on the fuel assembly contents having a specific configuration (e.g., geometric form, a certain number of fuel rods or solid replacement filler rods in the assembly lattice). Although the spent fuel assembly is not an SSC of the NAC-MPC System the spent fuel must remain in its analyzed configuration during the period of extended operation, for continuation of the approved design bases. Therefore, for the NAC-MPC System CoC renewal, the condition of the SNF assembly and cladding are within the scope of renewal and are reviewed for aging mechanisms and effects that may lead to a change in the analyzed fuel configuration.

The experimental confirmatory basis that low-burnup fuel (≤ 45 gigawatt days per metric ton of uranium (GWd/MTU)) will remain in its analyzed configuration during the period of extended operation was provided in NUREG/CR-6745, “Dry Cask Storage Characterization Project— Phase 1; CASTOR V/21 Cask Opening and Examination” [3.9.11], and NUREG/CR-6831, “Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage” [3.9.12]. This research demonstrated that low-burnup fuel cladding and other cask internals had no deleterious effects after 15 years of storage and confirmed the basis for the guidance on creep deformation and radial hydride reorientation in Interim Staff Guidance (ISG)-11, “Cladding Considerations for the Transportation and Storage of Spent Fuel, Revision 3” [3.9.10]. The NRC staff indicated, in ISG-11, Revision 3, that the spent fuel configuration is expected to be maintained as analyzed in the safety analyses for the NAC-MPC System, provided certain acceptance criteria (regarding maximum fuel clad temperature and thermal cycling) are met, and the fuel is stored in a dry inert atmosphere. The research results in NUREG/CR-6745 and NUREG/CR-6831 support the NRC staff’s determination that degradation of low-burnup fuel cladding and assembly hardware should not result in changes to the approved design bases during the first period of extended operation, provided that the TSC internal environment is maintained. The U.S. Department of Energy (DOE) gathered similar experimental confirmatory data to support the technical basis for storage of high-burnup (HBU) fuel during the first period of extended operation [3.9.290]. The NAC-MPC Systems loaded to date do not include any HBU fuel assemblies.

The staff reviewed gap assessments for dry storage systems, relevant technical literature, and operating experience from nuclear applications [3.9.20; 3.9.51; 3.9.85; 3.9.142; 3.9.129] to identify potential degradation mechanisms in consideration of the materials and condition of the SNF at loading and the environment in dry storage. The SNF cladding materials are zirconium-based or stainless-steel alloys. The primary components of the fuel assembly hardware are spacer grids, end fittings, guide tubes (PWR only), and assembly channels (BWR only). The materials of construction for these components include zirconium-based alloys, nickel alloys, and stainless-steel. The condition of the SNF assembly at loading considered changes to the fuel pellets and the zirconium-based and stainless-steel cladding during reactor service, including hydrogen absorption by the cladding, swelling of the fuel pellets, increased rod pressurization due to helium and fission gas release, and pellet-cladding interactions. The environment considered is helium cover gas in high radiation and temperature environment. A minimal amount of water (about 0.43-gram mole) is also considered to be retained inside the TSC [3.9.122]. This moisture content is based on a design-basis drying process that evacuates the TSC to less than or equal to 3 torr [0.06 psi] and backfills with high purity helium before closure.

The aging mechanisms considered for zirconium-based cladding include hydride-induced embrittlement, delayed hydride cracking, thermal and athermal (low-temperature) creep, localized mechanical overload, radiation embrittlement, fatigue, oxidation, pitting corrosion, galvanic corrosion, and SCC and MIC. Of these potential mechanisms, MIC was not considered to be applicable, as the aging mechanism is not expected to be operable under the inert atmosphere of dry storage. In addition, hydride-induced embrittlement and creep were not considered for low-burnup fuel, because confirmatory data were obtained in support of their disposition, as discussed previously. Detailed discussions regarding each of these applicable aging mechanisms for zirconium-based cladding are provided in Section 3.2.5.1.

Per the guidance of EPRI Report No. TR-106440 [3.9.353] the aging mechanisms considered for stainless steel clad SNF include general corrosion, stress corrosion cracking, localized corrosion (pitting), stress rupture, strain rate embrittlement, hydrogen-induced degradation, helium embrittlement, and fission product cladding interaction. Detailed discussions regarding each of these applicable aging mechanisms for stainless steel clad SNF are provided in Section 3.2.5.2.

The degradation mechanisms considered for the assembly hardware include creep, fatigue, hydriding, general corrosion, SCC, and radiation embrittlement. Detailed discussions regarding each of these applicable aging mechanisms for assembly hardware are provided in Section 3.2.5.3.

3.2.5.1 Cladding Materials – Zirconium Alloys

3.2.5.1.1 Hydride Reorientation and Hydride-Induced Embrittlement (High-Burnup [HBU] Fuel)

In reactor service, the zirconium-based fuel cladding absorbs hydrogen, which leads to the precipitation of hydride platelets as the dissolved hydrogen exceeds the solubility limit of the cladding. The primary source of the hydrogen is water-side corrosion (oxidation) of the cladding [3.9.85; 3.9.301]. The total concentration of hydrogen absorbed by the cladding (i.e., dissolved in the zirconium matrix and in precipitated hydrides) increases with burnup and varies axially across the fuel rods. For burnups above 45 GWd/MTU and up to 62 GWd/MTU (the current NRC licensing limit), the total hydrogen content for Zircaloy-2 is expected to be in the range of 260–300 weight parts per million [wppm], 200–1,200 wppm for Zircaloy-4, ≤ 100 wppm for M5®, and up to 550 ± 300 wppm for ZIRLO™.

The maximum allowable burnup of PWR and BWR SNF authorized contents in the NAC-MPC System is < 45 GWd/MTU, hydride reorientation is not credible during the 40-year period of extended operation. Therefore, significant hydride-induced embrittlement is also not considered a credible aging mechanism for NAC-MPC System SNF content claddings.

3.2.5.1.2 Delayed Hydride Cracking

Delayed hydride cracking (DHC) is a time-dependent mechanism traditionally thought to occur by the diffusion of hydrogen to an incipient crack tip (notch, flaw) in the cladding, followed by nucleation, growth, and subsequent fracture of the precipitated hydrides at the crack tip [3.9.85]. Hydrogen dissolved in the cladding can diffuse up a stress gradient in the crystalline lattice, or into the stress field at the core of an edge dislocation [3.9.284]. The concentration gradient established by the stress gradient may lead to hydrogen supersaturation (i.e., solubility limit being exceeded) leading to the precipitation of hydrides at the crack tip. The precipitated hydride will continue to grow by the dissolution of hydrides in the low-stress regions of the material and by the continued diffusion of hydrogen up the stress gradient. Once the hydride reaches a critical size, it will crack and propagate to the end of the hydride, where it will blunt. The cycle could then repeat, until the crack propagates through the thickness of the material. DHC of spent fuel cladding has been studied under thermal transients representative of reactor operation [3.9.315; 3.9.310] and representative of dry storage [3.9.335; 3.9.352].

Requisite conditions for DHC are the presence of: (i) hydrides, (ii) existing crack tips (notch, flaws) that act as initiating sites, and (iii) sufficient cladding hoop stresses. Simpson and Ells [3.9.340] observed DHC with hydrogen concentration as little as 10 ppm in Zr-2.5 percent Nb cladding, although testing was performed at room temperature (i.e., a much lower temperature than those expected during the renewal period). Similarly, Coleman et al. [3.9.288] were able to induce DHC in Zircaloy-4 at 200 wppm of hydrogen. Regarding requisite existing (incipient) crack tips, EPRI estimated the maximum initial depth of existing crack tips to be 140 μm [5.5 mils] or approximately 28 percent of the remaining wall of a typical 17x17 PWR cladding with 600 μm [23.6 mils] of original cladding thickness, and 100 μm [4 mils] of oxidation during its exposure in the reactor. Conversely, Raynaud and Einziger [3.9.329] estimated the maximum initial depth of existing crack tips to be 120 μm [4.7 mils] for a cladding oxide thickness of 100 μm [4 mils]. Regarding requisite hoop stresses for crack initiation, the mechanism requires that the stress intensity factor at the crack tip exceed a threshold value, denoted as K_{IH}.

Most DHC studies have been performed under thermal transients representative of reactor operation, primarily on CANDU pressure tubes (Zr–2.5 percent Nb) and Zircaloy-2 cladding. Chan [3.9.281] conducted an extensive literature review of experimentally determined K_{IH} values for DHC crack initiation. In that review, K_{IH} values for Zircaloy-2 are in the range of 5–14 MPa√m [4.55–12.74 ksi√in] at 25°C – 300°C [77°F – 572°F], and in the range of 5–10 MPa√m [4.55–9.10 ksi√in] for Zr-2.5 percent Nb cladding at 75°C – 300°C [167°F – 572°F] [3.9.281, Figures 2 and 3]. Kubo et al. [3.9.315] also compiled K_{IH} values for Zircaloy-2 in the range of 3–13 MPa√m [2.73–11.8 ksi√in]. Kim [3.9.309] also measured a K_{IH} value of 2.5 MPa√m [2.28 ksi√in] for Zr-2.5 Nb cladding at 160°C [320°F]. Based on the available data, the staff considered a reference K_{IH} value of 5.0 MPa√m [2.73 ksi√in] for comparison with requisite stress intensity factors or minimum flaw sizes for DHC initiation.

Raynaud and Einziger [3.9.329] estimated the cladding hoop stresses while conservatively accounting for release of fission gases and decay gases during storage, including stresses due to radiation-induced pellet swelling during storage. Raynaud and Einziger concluded that DHC cannot occur for a K_{IH} of 5 MPa√m [4.55 ksi√in], because the flaw size needed to induce DHC is much larger than the initial depth of potential existing cracks (120 μm [4.7 mils]). The estimated critical flaw size needed to initiate DHC in BWR fuel cladding is larger than 50 percent of the cladding thickness for 300 years of dry storage. For PWR cladding, the critical flaw size is larger than 30 percent of the cladding thickness for the first 5 years of the dry storage and larger than 50 percent of the cladding thickness beyond the first 5 years up to 300 years of dry storage. The calculations for the hoop stresses in ZIRLO™-clad IFBA rods with hollow and solid blanket pellets, which are expected to be higher than standard rods show that the critical flaw size for the PWR cladding is still larger than 30 percent of the cladding thickness for the first 5 years of dry storage and larger than approximately 45 percent of the cladding thickness beyond the first 5 years up to 300 years of dry storage. Therefore, it is concluded that the critical flaw size needed to induce DHC, in both standard and IFBA rods, is much larger than the initial depth of potentially existing cracks (120 µm [4.7 mils]). As NAC-MPC System cladding temperatures are below design-bases peak cladding temperature will be below the limits defined in ISG-11, Revision 3 (i.e., 400°C [752°F]) in storage during the period of extended operation resulting in decreased cladding hoop stresses.

Based on the NRC staff analysis in MAPS, it has been determined that significant DHC is not a credible aging mechanism for the NAC-MPC System during the 40-year period of extended operation, and therefore, aging management is not required

3.2.5.1.3 Thermal Creep (High-Burnup [HBU] Fuel)

Creep is the time-dependent deformation of a material under stress. Creep in zirconium-based cladding is caused by the hoop stresses from the rod internal pressure at a given fuel temperature; it is expected to be self-limiting, due to the decreasing temperatures and creep-induced volume expansion, which results in lower internal rod pressures with time. Excessive creep of the cladding during dry storage could lead to thinning, hairline cracks, or gross ruptures [3.9.85], which may affect the ability to safely retrieve the HBU fuel on a single-assembly basis (if required by the design bases).

The maximum allowable burnup of PWR and BWR SNF authorized contents in the NAC-MPC System is < 45 GWd/MTU, thermal creep of zirconium-based cladding is not credible during the 40-year period of extended operation. Therefore, significant thermal creep of zirconium-based cladding is also not considered a credible aging mechanism for NAC-MPC System SNF content zirconium alloy cladding.

3.2.5.1.4 Low-Temperature Creep

Low-temperature creep (also called "athermal creep") may occur when sustained hoop stresses operate on the cladding material at or near ambient temperature [3.9.20]. Various athermal creep mechanisms have been proposed at low stresses (e.g., Nabarro-Herring, Coble, and Harper-Dorn creep mechanisms) [3.9.323], although there is no evidence or literature information to support that these will be operational on zirconium-based alloys. However, the literature shows that low-temperature creep has been shown to occur in titanium and its alloys, which leads to deformation twinning [3.9.305]. Since both titanium and zirconium have the same crystalline structure (hexagonal close packed crystalline), the zirconium-based cladding was reviewed for its susceptibility to low-temperature creep.

In materials such as α and α-β titanium alloys, which are comparable to the zirconium-based alloys used for fuel cladding, low-temperature creep has been observed when tensile stresses exceed 25 percent of the yield strength [3.9.275]. For example, Ankem and Wilt reported a threshold stress in the range of 25–50 percent of the yield stress for Ti Grade 7, and 35–60 percent of the yield stress for Ti Grade 24. The yield strength of the irradiated zirconium-based cladding at low temperatures (550–1,000 MPa [79.8–145 ksi]; [3.9.297; 3.9.293; 3.9.280] is expected to be close to the yield strength of Ti Grade 24 (825 MPa [119.6 ksi]) and well above the yield strength of Ti Grade 7 (275 MPa [39.9 ksi]) [3.9.302]. Therefore, the staff considered the results in Ankem and Wilt to provide reasonable acceptance criteria for determining if low-temperature creep is a credible aging mechanism in the 60-year time frame.

The main sources of sustained hoop stresses at low temperatures are expected to be the rod internal pressure and pellet-cladding mechanical interaction (PCMI). Raynaud and Einziger [3.9.329] estimated the cladding hoop stresses after 300 years of storage to be approximately 25 MPa [3.62 ksi] and 35 MPa [5.07 ksi] for representative BWR and PWR fuel cladding, respectively. These estimates accounted for a credible release of fission and decay gases to the fuel-cladding interspace, pellet swelling, and fuel and cladding temperature. The hoop stresses for IFBA rods are conservatively expected to be around or less than 75 MPa [10.87 ksi] [3.9.279]. These hoop stress estimates are all less than 25 percent of the yield strength of zirconium-based cladding, i.e., below the expected range of 550–1,000 MPa [79.8–145 ksi] near ambient temperature for cladding with circumferential hydrides only [3.9.297; 3.9.293; 3.9.280]. Further, more recent data [3.9.312; 3.9.313] suggest that, even with the potential decrease in yield strength due to radial hydrides (which conservatively does not account for a potential increase in yield strength due to irradiation), the hoop stresses in the cladding are still maintained below 25 percent of the yield strength of irradiated cladding with both circumferential and radial hydrides.

Raynaud and Einziger acknowledged that the low-temperature creep models are not programmed into FRAPCON-DATING, which the authors used to predict the elevated temperature cladding creep (see Section 3.2.5.1.3). The authors noted that extrapolations of the high-temperature cladding creep model results in immeasurably small values of cladding strains at low temperature. However, the lack of cladding creep beyond 50 years (corresponding to temperatures below approximately 200°C [392°F]) results in smaller strains being predicted in these calculations. Therefore, the calculated cladding hoop stresses are conservative when compared to the 25-percent criteria, as athermal creep-induced strains would reduce these stresses.

The previously discussed Raynaud and Einziger study did not account for potential stress concentration effects due to pellet-pellet interfaces and pellet fragment-to-fragment friction forces that could result in more severe PCMI than for a perfectly cylindrical pellet (as assumed in the paper). Recently, Ahn et al. [3.9.274] estimated stress concentrations from pellet-clad mechanical stresses caused by the radiation-induced pellet swelling up to 100 years, independent of hoop stresses due to fission and decay gas release. The work estimated that, for HBU fuel, the average pellet-swelling-induced PCMI stress concentration was on the order of 200 MPa [29 ksi] locally. Literature indicates that radiation-induced pellet swelling is expected to reach its maximum value beyond the 60-year timeframe [3.9.331; 3.9.332; 3.9.333]. Therefore, PCMI stress concentrations due to radiation-induced pellet swelling are not expected to exceed a threshold stress of 25 percent of the yield stress (similar to the titanium data in 3.9.275) during the 60-year timeframe.

In summary, literature on the creep strain and creep rate of the zirconium-based cladding materials at room temperature per the hoop stresses expected during extended storage is not available. Therefore, it is not possible to directly assess the low-temperature creep of the zirconium-based cladding materials. However, the threshold levels of tensile stresses for low-temperature creep in the similar crystalline-structured (hexagonal close packed crystalline) materials, which indicate that cladding hoop stresses on the cladding must exceed approximately 25 percent of yield strength for athermal creep to be credible. The room temperature hoop stresses on the zirconium-based cladding are expected to be less than 25 percent of the yield strength. Therefore, the low-temperature (athermal) creep mechanism is not considered credible, even for the unlikely scenario where fuel reaches room temperature during the 40-year period of extended operation. Therefore, aging management for the NAC-MPC System for low-temperature creep is not required during the 40-year period of extended operation.

3.2.5.1.5 Mechanical Overload

Mechanical overload is generally associated with pellet-to-cladding interaction (PCMI), which could compromise the cladding integrity during storage. PCMI is likely during reactor operations when the reactivity transient during a reactivity-initiated accident (RIA) results in a rapid increase in a fuel rod power, leading to a nearly adiabatic heating of the fuel pellets and potential failure of the fuel cladding. In either commercial BWRs or PWRs, cladding failures have not been attributed to PCMI. However, data generated in experimental reactors conducting ramp testing of heavily hydrided fuel claddings indicate that hydride rims with large hydride number density at the cladding outer surface may lead to crack initiation [3.9.273]. The cracks could propagate from the outside toward the inner cladding surface, potentially resulting in failures.

During dry storage, PCMI stresses could develop due to pellet swelling and release of fission gases to the gap between the fuel and cladding. PCMI could lead to the opening of existing flaws in the cladding, potentially resulting in the release of fission gases and other fission products into the cask environment. The existing flaws in undamaged fuel are likely to be of any of the following: (i) surface (non-through-wall) cracks on the inner or outer wall; (ii) hairline cracks; (iii) wall thinning due to oxide spallation on the outer surface; or (iv) wall thinning due to fretting wear on the outer surface [3.9.20].

Due to low levels of creep strain, strain rate, and temperature-dependent hoop stresses experienced during NAC-MPC System dry storage operations, it is concluded that cladding failures due to PCMI-induced mechanical overload are not considered credible during the 40-year period of extended operation, and aging management is not required.

3.2.5.1.6 Oxidation

In the presence of residual amounts of water and high enough temperature, zirconium-based cladding can be oxidized according to the following chemical reaction: $Zr + 2H_2O = ZrO_2 + 2H_2$ [3.9.307; 3.9.284; 3.9.334]. Various scoping calculations were performed [3.9.307] to determine the extent of cladding oxidation during dry storage in the presence of up to 1 L [0.26 gal] (equivalent to 55.5 moles) of residual water. The amount of residual water considered is significantly higher than the residual water amount of 0.43 moles expected after vacuum drying.

The scoping calculations were based on a representative storage system loaded with the equivalent of 21 Babcock & Wilcox SNF assemblies, each containing 208 fuel rods in a storage canister. It was concluded that the maximum cladding thickness loss due to temperature-dependent cladding oxidation kinetics for both Zircaloy-2 and Zircaloy-4 is not expected to exceed 10 μm [0.4 mils], even with complete consumption of the assumed 1 L [0.26 gal] of residual water. The loss of cladding thickness due to oxidation represents less than 2 percent of the original cladding thickness. Therefore, cladding oxidation is considered to be insignificant, and aging management for cladding oxidation in the NAC-MPC System is not required during the 40-year period of extended operation.

3.2.5.1.7 Pitting Corrosion

Pitting corrosion initiates and propagates when (i) there is an aggressive chemical environment that results in corrosion potential being greater than the repassivation potential and (ii) there is enough cathodic capacity to sustain the propagation of the pitting corrosion [3.9.338]. Zirconium is a passive material and is protected by a ZrO_2 surface film [3.9.328]. The surface oxide readily reforms if broken, but zirconium is not completely immune to pitting as halides (i.e., anions of fluorine, chlorine, bromine, and iodine) in aqueous or gaseous forms could initiate pitting.

Inside the NAC-MPC System TSC's internal environment, a limited amount of residual water is expected to be retained following drying, which will be in the liquid state once temperatures are near or below 100°C [212°F]. The residual water amount is expected to be less than 1 mole per NUREG-1536 [3.9.122]. During storage, most residual water is expected to decompose into hydrogen and oxidizing species, such as oxygen and hydrogen peroxide [3.9.307]. It is possible for trace amounts of water to remain in the vapor phase but is not expected to be in the liquid phase during dry storage, due to the low relative humidity in the TSC cavity. The relative humidity inside the NAC-MPC System TSC cavity assuming a residual water content of 0.43 mole at 25°C [77°F], is estimated to be approximately 15 percent using a helium backfill pressure of 1 atmosphere (atm) [14.7 psi]. Any residual water in the vapor phase is expected to be spread throughout the TSC cavity and is not expected to be sufficient to provide enough cathodic capacity to initiate and propagate pitting corrosion of the cladding. Therefore, pitting corrosion of the cladding of fuel assemblies stored in the NAC-MPC System is not considered credible, and aging management is not required during the 40-year period of extended operation.

3.2.5.1.8 Galvanic Corrosion

Galvanic corrosion can occur due to a mismatch in corrosion potentials between two metals in an aqueous solution. In fuel assemblies, the mismatch can occur when the cladding is in contact with other metallic components, which could result in the formation in a galvanic cell, provided there is an aqueous solution between the two subcomponents. For example, some of the PWR and BWR fuel assemblies contain spacer grids that are made of Inconel alloys, such as Inconel 718 and Inconel 625. The dominant constituents of these Inconel alloys include nickel, chromium, molybdenum, iron, niobium, and tantalum. A galvanic cell could form if residual water condenses in the gap between the rod and a spacer grid, simultaneously contacting both materials. The cladding could also be covered with a crud layer deposit during reactor operations, which could further facilitate formation of the contact.

The amount of residual water inside the TSC following drying is expected to be less than 1 mole after vacuum drying. Most residual water is expected to decompose over time into hydrogen and oxidizing species, such as oxygen and hydrogen peroxide. It is possible for some trace amount of water to remain in the vapor phase inside the canister after the first renewal period but is not expected to condense into liquid phase during dry storage due to the low relative humidity of the containment cavity. Further, any residual water in the vapor phase is expected to be spread throughout the containment cavity and is not expected to be sufficient to form a corrosion cell between the cladding and the spacer grids made of Inconel alloys. Therefore, galvanic corrosion of the zirconium-based cladding alloys of spent fuel assemblies stored in the NAC-MPC System is not considered credible, and aging management is not required during the 40-year period of extended operation.

3.2.5.1.9 Stress-Corrosion Cracking

SCC occurs as a result of a synergistic combination of a susceptible material, an aggressive environment, and sufficiently high tensile stress. The corrosive environment associated with SCC of fuel rods has been attributed to specific fission products, such as iodine, cesium, and cadmium, generated during reactor irradiation [3.9.348; 3.9.339]. SCC of the cladding can occur at the rod's inner surface where the fuel pellet and cladding mechanically interact and is related to PCMI hoop stresses on the cladding. SCC of zirconium-based cladding has been observed in BWRs during power ramp-up [3.9.327; 3.9.273]. PWR cladding is unlikely to undergo similar SCC because of the more gradual power ramp-up. Fuel pellets in PWR cladding are unlikely to undergo sudden expansion and induce high stresses, as in BWR cladding. No cladding failures from SCC are known to have occurred either during pool storage or under dry storage conditions.

Even with the PCMI-induced hoop stresses, the cladding stresses will remain well below the 240 MPa [34.8 ksi] criterion for inducing SCC. Therefore, SCC of the cladding of fuel assemblies stored in the NAC-MPC System is not considered credible, and aging management is not required during the 40-year period of extended operation.

3.2.5.1.10 Radiation Embrittlement

Radiation embrittlement of cladding can result in degradation of the mechanical properties of the cladding, such as ductility and strength. This can lead to the reduction in the maximum load that the cladding can withstand, potentially leaving the cladding vulnerable to failure under external loads. Because radiation embrittlement is associated with a cumulative fluence of on the order of 10^{22} n/cm^2, which is not expected during NAC-MPC System dry storage operations, radiation embrittlement of cladding is not considered credible, and therefore, aging management is not required during the 40-year period of extended operation.

3.2.5.1.11 Fatigue

Fatigue occurs when a material is subjected to repeated loading and unloading stresses. If the loads are above a certain threshold, microscopic cracks will begin to form at stress concentrators at the surface, persistent slip bands, and grain interfaces. As a crack reaches a critical size, it will propagate until fracture. Because dry storage is a passive application, purely mechanical cyclic loading is not expected. However, the cladding will experience thermal cycles due to daily and seasonal fluctuations in ambient temperature, as well as extreme weather events within a larger seasonal pattern. These thermal cycles will induce cyclic stresses on the cladding due to either (i) changes in fission and decay gas pressure, as governed by gas laws, which would result in fluctuations in cladding hoop stresses, and (ii) partial restraint on cladding thermal expansion and contraction due to top and bottom nozzles, hold-down springs, and spacer grids. These thermally induced stresses and corresponding strains can produce fatigue damage in the same manner as purely mechanical cyclic loading.

Steady-state analyses conducted [3.9.289] show that the change in peak cladding temperature is directly proportional to the change in external air temperature of the TSC. Although the large thermal mass of the NAC-MPC System is likely to reduce the amplitude and frequency of the thermal cycles on fuel and cladding temperature, even a correlation coefficient of unity between the peak cladding and external air temperature does not result in excessive cladding hoop stresses. In conclusion, the cumulative cyclic stresses for all daily and seasonal temperature cycles result in stresses ranging from 20 to 70 MPa [2.9 and 10.2 ksi] for BWR and from 65 to 115 MPa [9.4 and 16.7 ksi] for PWR claddings. Even the combined conservative values are well below the threshold of 260 MPa [37.7 ksi] needed for fatigue-induced failure in the cladding. Therefore, fatigue-induced failure of the cladding is not credible in the NAC-MPC System during the 40-year period of extended operation, and aging management is not required.

3.2.5.2 Cladding Materials – Stainless Steel Alloys

3.2.5.2.1 *General Corrosion*

General corrosion of the stainless steel (SS) cladding of SNF could reduce the wall thickness and could enhance the possibility of degradation from other mechanisms. Per information contained in EPRI Report No. TR-106440, “Evaluation of Expected Behavior of LWR Stainless Steel-Clad Duel in Long-Term storage” [3.9.353], reported that investigations into corrosion of SS-SNF indicates that general corrosion of the cladding during dry storage should not be of concern when the TSC is vacuum dried following lid closure welding and backfilled with a high-purity (> 99.9%) inert helium atmosphere.

Based on reported research, general corrosion is not a credible aging mechanism for the 40-year period of extended operation, and therefore aging management is not required.

3.2.5.2.2 *Stress Corrosion Cracking*

Per EPRI Report [3.9.353], thermal sensitization of SS cladding is limited as the storage temperatures for the SS-SNF are expected to be less than the temperatures required for thermal sensitization (> 800-degree F). Therefore, SCC due to sensitization is not expected to promote significant cladding degradation with the SS-SNF stored in TSCs which have been vacuum dried and backfilled with a high-purity (> 99.9%) inert helium atmosphere.

Based on reported research, stress corrosion cracking is not a credible aging mechanism for the 40-year period of extended operation, and therefore aging management is not required.

3.2.5.2.3 *Localized Corrosion (Pitting)*

Pitting attack is not expected to be a significant issue for SS-SNF stored in a vacuum dried and high-purity (> 99.9%) inert helium atmosphere of a closed and welded TSC [3.9.353].

Based on reported research, pitting corrosion is not a credible aging mechanism for the 40-year period of extended operation, and therefore aging management is not required.

3.2.5.2.4 *Stress Rupture*

Rapid stress rupture requires hoop stresses in excess of the yield stress, approximately 210 MPa at 400 degrees C [3.9.356]. This mechanism is not generally of concern during dry storage because there are no credible sources of primary stresses sufficiently high to generate this type of cladding breach in intact SNF during storage.

Even though crack-free cladding may not be susceptible to rapid stress rupture, high localized stresses may develop in regions of incipient cladding defects that are formed during irradiation. Therefore, fuel rods with incipient cladding failures have a higher susceptibility to cladding breach during dry storage. Propagation of an incipient crack to a small pin hole vents the internal gas pressure to the storage system. Venting of the fission gas relieves the pressure and terminates the cracking process. Releases of Kr^{85} from Zircaloy-clad SNF during dry storage tests at Nevada Test Site, GE Morris and Idaho National Engineering Laboratory were attributed to opening of pin-hole breaches at sites of incipient defects [3.9.363; 3.9.362, 3.9.357].

Based on reported research, stress rupture is not a credible aging mechanism for the 40-year period of extended operation, and therefore aging management is not required.

3.2.5.2.5 *Strain Rate Embrittlement*

Strain rate embrittlement and triple point cracking occur at stresses much higher than those anticipated during dry storage operations. Based on fracture maps for SS 316 [3.9.358], maximum shear stresses higher than 105 MPa (210 MPa maximum hoop stress) would be required for this failure mode, whereas maximum shear stresses estimated for SS-SNF are only 75 MPa (150 MPa maximum hoop stress).

Based on reported research, strain rate embrittlement is not a credible aging mechanism for the 40-year period of extended operation, and therefore aging management is not required.

3.2.5.2.6 *Hydrogen-Induced Degradation*

In contrast to Zircaloy clad fuel, research indicates that hydrogen embrittlement of SS is not an issue because hydrogen has low solubility and high mobility in SS. Therefore, hydrogen concentrations should be low (< 1 ppm) and have little impact on the storage behavior of SS-SNF [3.9.361, 3.9.359].

Based on reported research, hydrogen-induced degradation is not a credible aging mechanism for the 40-year period of extended operation, and therefore aging management is not required.

3.2.5.2.7 *Helium Embrittlement*

Helium is generated in SS in thermal reactors by reactions with boron, nitrogen, and certain reaction products of nickel. For bounding LWR SS-SNF assembly average and assembly peak fast neutron fluences of 1.0 and 1.2 x 10^{22} n/cm^2 (E > 1 MeV) and assembly average and assembly peak total neutron fluences of 3 and 4 x 10^{22} n/cm^2 (E > 10^{-10} MeV), the helium content could be in excess of 100-200 ppm [3.9.364]. The helium is mobile above 400 degree C [3.9.353].

Based on reported research, lower burnups and longer cooling times of SS-SNF, helium embrittlement is not a credible aging mechanism for the 40-year period of extended operation, and therefore aging management is not required.

3.2.5.2.8 *Fission Product Cladding Interaction*

Fission product SS cladding interaction has not been noted [3.9.363] and SS-clad fuel has not appeared to be susceptible to SS-SNF in dry storage conditions.

Based on reported research, failures due to fission product cladding interaction are not expected during dry storage because of the absence of thermal cycling and high stresses, and therefore, fission product cladding interaction is not a credible aging mechanism for the 40-year period of extended operation and aging management is not required.

3.2.5.2.9 *Fatigue*

Fatigue occurs when a material is subjected to repeated loading and unloading stresses. If the loads are above a certain threshold, microscopic cracks will begin to form at stress concentrators at the surface, persistent slip bands, and grain interfaces. As a crack reaches a critical size, it will propagate until fracture. Because dry storage is a passive application, purely mechanical cyclic loading is not expected. However, the cladding will experience thermal cycles due to daily and seasonal fluctuations in ambient temperature, as well as extreme weather events within a larger seasonal pattern. These thermal cycles will induce cyclic stresses on the cladding due to either (i) changes in fuel rod internal gas pressure, as governed by gas laws, which would result in fluctuations in cladding hoop stresses, and (ii) partial restraint on cladding thermal expansion and contraction due to top and bottom nozzles, hold-down springs, and spacer grids. These thermally induced stresses and corresponding strains can produce fatigue damage in the same manner as purely mechanical cyclic loading.

A fatigue evaluation of the stainless-steel cladding for the Connecticut Yankee, Yankee Rowe, and LACBWR while in long term dry storage has been performed [3.9.366]. The calculation is based on the fatigue evaluation for zirconium-based cladding outlined in NUREG-2214 [3.9.4] and the information contained in EPRI Report No. TR-106440 [3.9.353]. No external loading mechanism to apply mechanical cycling to the fuel in dry storage is expected and only thermal cycling is considered. Daily 45°F and seasonal 257°F cyclic changes in the ambient air temperature are used in the evaluation. The ambient air temperature fluctuations are applied directly to the fuel cladding and a time span of 60 years is considered. In addition, the evaluation includes stresses developed from monotonic cooling of the cladding from a bounding temperature at maximum heat load 716°F (380°C) when placed into dry storage down to an ambient temperature of -40°F.

The alternating circumferential and radial stresses in the cladding from the daily, seasonal, and cooling to ambient conditions are calculated by treating the cladding as a thin cylinder. Using these stresses, the fatigue evaluation follows the methodology in ASME B&PVC Section III, Div. 1, Subsection NB, and considers the cumulative damage from the different variations in the magnitudes of the cyclic temperatures, associated number of cycles, and the resulting alternating stresses. The calculated value of the cumulative usage factor is 0.005 for the Connecticut Yankee, Yankee Rowe, and LACBWR fuel cladding. This is significantly less than the acceptance criterion of 1.0 and indicates a large margin against fatigue failure.

Therefore, fatigue-induced failure of the stainless-steel cladding is not credible in the NAC-MPC System during the 40-year period of extended operation, and aging management is not required.

3.2.5.3 Assembly Hardware Materials

The assembly hardware considered here includes guide tubes, spacer grids, and lower and upper end fittings. The guide tubes are fabricated using zirconium-based alloys. The other components are fabricated using one of the following materials: zirconium-based alloys, Inconel 718, Inconel 625, Inconel X-750, and stainless steel 304L. These subcomponents are not expected to experience sustained external loads during passive dry storage except for their own weight.

Based on an evaluation of the analysis of the MAPS draft report [3.9.4], there are no credible aging mechanisms such as creep, SCC, fatigue, hydriding, general corrosion or radiation embrittlement that will significantly affect the performance of the SNF assembly hardware stored in the NAC-MPC System during the 40-year period of extended operation, and therefore, aging management is not required.

Table 3.2-1 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent [1]	Material [3]	Storage Operation Environment [4]	Aging Mechanism	Aging Effect	Aging Management Activities
455-870-1	Shell	Stainless Steel	HE	Radiation Embrittlement	Cracking	No
			SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Fatigue	Cracking	TLAA per Design Code
				Microbiologically Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
		Stainless Steel (Welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
455-870-2	Bottom	Stainless Steel	HE	Radiation Embrittlement	Cracking	No
			SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Fatigue	Cracking	TLAA per Design Code
				Microbiologically Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
		Stainless Steel (Welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP

Table 3.2-1 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
455-871-3, -9	Shield Lid / Shield Lid – Damaged Fuel	Stainless Steel	FE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
			HE	Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	FE	Stress Corrosion Cracking	Cracking	No
455-871-5	Structural Lid	Stainless Steel	SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Fatigue	Cracking	TLAA per Design Code
				Microbiologically Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	FE	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
455-871-2	Spacer Ring	Stainless steel (welded)	FE	Stress Corrosion Cracking	Cracking	No
		Stainless steel	FE	Pitting and Crevice Corrosion	Loss of material (Precursor to stress corrosion cracking)	No
				Microbiologically Influenced Corrosion	Loss of material	No
				Radiation Embrittlement	Cracking	No

Table 3.2-1 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
455-871-7	Port Cover	Stainless steel	FE	Creep	Change in Dimensions	No
				Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
			SH	Radiation Embrittlement	Cracking	No
		Stainless Steel (Welded)	FE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
455-871-1	Shield Lid Support Ring	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
		Stainless Steel (Welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
455-881- 1, -3, -4, -5, -6	PWR Fuel Tube, Cladding, Flange	Stainless Steel (Welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
455-881-2	Neutron Absorber	Boral	HE	Boron Depletion	Loss of Criticality Control	TLAA
				General Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Thermal Aging	Loss of Strength	No
				Wet Corrosion and Blistering	Change in Dimensions	No
				Galvanic Corrosion	Loss of Material	No
				Creep	Change in Dimensions	No

Table 3.2-1 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(3)]	Storage Operation Environment [(4)]	Aging Mechanism	Aging Effect	Aging Management Activities
455-891-1, -2, -3, -4, -5, -6	Bottom Fuel Basket Weldment	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
455-892-1, -2, -3, -4, -5, and 455-895- 16, -17	Top Fuel Basket Weldment	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
455-893-1	Fuel Basket Support Disk	Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
				General Corrosion	Loss of Material	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel (17-4 PH)	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No

Table 3.2-1 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
455-893- 2, -3, -5, -6, -7	Fuel Basket Tie Rods, Spacers, and Washers	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel (Welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
455-893-4	Fuel Basket Top Nut	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
				Stress Relaxation	Loss of Preload	No
455-894-1	Fuel Basket Heat Transfer Disk	Aluminum	HE	Thermal Aging	Loss of Strength	No
				General Corrosion	Loss of Material	No
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
455-895-13	Fuel Basket Flat Washer	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No

Table 3.2-1 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
455-902-1,-4, -6, -7, -14, -15	Damaged Fuel Can (DFC) Screen Cover Plate, and Filter and Backing Screens	Stainless Steel	HE	Creep	Change in Dimensions	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
				Radiation Embrittlement	Cracking	No
455-902-2, - 5	DFC Lid Plate and Bottom Plate	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
455-902-8	DFC Bottom and Side Plates	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
455-902-9	DFC Lid Collar Upper Side Plates	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No

Table 3.2-1 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
455-902-10	DFC Tube Body	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
455-902-12, -13, -16	DFC Lift Tee, Support Ring and Dowel Pin	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	HE	Creep	Change in Dimensions	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
455-919-1, -2, -3, -4, -5	Test Assembly Retainer Lower Tab, Lifting Plate, Gusset and Ring	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No
YR-00-061-1, - 2	YR-RFA (2) Casing and Top Ring	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No
YR-00-062, Sheet 1. -2, -3.-4 Sheet 2, -1 Sheet 3, -10,	RFA Cylinder, inside Tab, Outside Tab, Top End Plate & Template	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No
YR-00-063-1, -2, -3, -4, -5	RFA Bottom End Fitting Assembly	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No
YR-00-064-1, -5	RFA Captive Bolt and Alignment Pin	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No
YR-00-065-1, -2	RFA Corner Angle & Tie Plate	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No

Table 3.2-1 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
YR-00-066-1, -2, -3	RFA Fuel Tube, Top Cap & Bottom Cap	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No

Notes

(1) Safety functions and item/note numbers of YR-MPC TSC and Fuel Basket Subcomponents are identified in Table 2.5-1.

(2) Yankee-Class Reconfigured Fuel Assembly (YR-RFA)

(3) Materials Legend: Steel = Carbon Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless Steel and Stainless Steel (welded) (including precipitation hardened stainless steel); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS3); Boral = Borated aluminum-based composites; Concrete; and Spent Nuclear Fuel.

(4) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas).

Table 3.2-2 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
414-870-1	Shell	Stainless Steel	HE	Radiation Embrittlement	Cracking	No
			SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Microbiologically Influenced Corrosion	Loss of Material	No
				Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
414-870-2	Bottom	Stainless Steel	HE	Radiation Embrittlement	Cracking	No
			SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Microbiologically Influenced Corrosion	Loss of Material	No
				Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
414-871-3	Shield Lid	Stainless Steel	HE	Radiation Embrittlement	Cracking	No
			FE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	FE	Stress Corrosion Cracking	Cracking	No

Table 3.2-2 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
414-871-5	Structural Lid	Stainless Steel	SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Fatigue	Cracking	TLAA per Design Code
				Microbiologically Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	FE	Stress Corrosion Cracking	Cracking	No
			SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
414-871-2	Spacer Ring	Stainless steel (welded)	FE	Stress corrosion cracking	Cracking	No
		Stainless steel	FE	Pitting and crevice corrosion	Loss of material (Precursor to stress corrosion cracking)	No
				Microbiologically influenced corrosion	Loss of material	No
				Radiation embrittlement	Cracking	No

Table 3.2-2 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
414-871-7	Port Cover	Stainless Steel	FE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Microbiologically influenced corrosion	Loss of material	No
				Radiation Embrittlement	Cracking	No
			SH	Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	FE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
414-871-1	Shield Lid Support Ring	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
414-881-1, -3, -4, and 414-882-1, -3, -4	Fuel Tube (Standard and Oversize), Cladding and Flange	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
414-881-2 and 414-882-2	Neutron Absorber	Boral	HE	Boron Depletion	Loss of Criticality Control	TLAA
				General Corrosion	Loss of Material	No
				Thermal Aging	Loss of Strength	No
				Wet Corrosion and Blistering	Change in Dimensions	No
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
				Galvanic Corrosion	Loss of Material	No

Table 3.2-2 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(3)]	Storage Operation Environment [(4)]	Aging Mechanism	Aging Effect	Aging Management Activities
414-891-1, -2, -3, -4, -5, -6	Fuel Basket Bottom Weldment (Welded)	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
414-892-1, -2, -3, -4, -5, -6, -7	Fuel Basket Top Weldment (Welded)	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
414-893-1	Fuel Basket Support Disk	Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
				General Corrosion	Loss of Material	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel (17-4 PH)	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No

Table 3.2-2 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
414-893-2, -3, -5, -6, -7	Fuel Basket Spacers and Tie Rods, Washers	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
414-893-4	Fuel Basket Top Nut	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
				Stress Relaxation	Loss of Preload	No
414-894-1	Fuel Basket Heat Transfer Disk	Aluminum	HE	Thermal Aging	Loss of Strength	No
				General Corrosion	Loss of Material	No
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
414-902-2, -5, -16	DFC Lid and Bottom Plates, and Dowel Pin	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No

Table 3.2-2 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(3)]	Storage Operation Environment [(4)]	Aging Mechanism	Aging Effect	Aging Management Activities
414-902-1. -8, -13	DFC Collar, Side and Bottom Plates	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
414-902-9	DFC Tube Body	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
414-902-11, -12	DFC Lift Tee and Support Ring	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
414-902--4, -6, -7, -14, -15	DFC Wiper, and Filter and Backing Screen	Stainless Steel	HE	Creep	Change in Dimensions	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
				Radiation Embrittlement	Cracking	No
414-903-4, -5, -8, -9, -10, -16, -17	CY-RFA [(2)] Corner Angle, Tube, Screens, Pin, Bolt & Support Grid	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No

Table 3.2-2 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (3)	Storage Operation Environment (4)	Aging Mechanism	Aging Effect	Aging Management Activities
414-904-1, -2, -3, -4, -5, -6, -7, -8	RFA Bottom Housing, Retaining Plate & Ring, Top Housing, Guide & Retaining Plate, Screen Ring & Housing	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No

Notes:

(1) Safety functions and item/note numbers of Concrete Cask Subcomponents are identified in Table 2.5-2.
(2) CY Reconfigured Fuel Assembly (RFA)
(3) Materials Legend: Steel = Carbon Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless steel (including precipitation hardened SS); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS-3); Boral = Borated aluminum-based composites; Concrete; and Spent Nuclear Fuel.
(4) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas).

Table 3.2-3 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
630045-870-1	Shell	Stainless Steel	HE	Radiation Embrittlement	Cracking	No
			SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Microbiologically Influenced Corrosion	Loss of Material	No
				Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
		Stainless Steel (Welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
630045-870-2	Bottom Plate	Stainless Steel	HE	Radiation Embrittlement	Cracking	No
			SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Microbiologically Influenced Corrosion	Loss of Material	No
				Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP

Table 3.2-3 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment [(3)]	Aging Mechanism	Aging Effect	Aging Management Activities
630045-871-1	Closure Lid	Stainless Steel	HE	Radiation Embrittlement	Cracking	No
			SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Microbiologically Influenced Corrosion	Loss of Material	No
				Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
630045-870-4	Lid Support Ring	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
630045-870-5	Inner Port Cover	Stainless Steel	FE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
			SH	Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	FE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No

Table 3.2-3 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
630045-870-7	Closure Ring	Stainless Steel	SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Fatigue	Cracking	TLAA per Design Code
				Microbiologically Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
			FE	Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	SH	Radiation Embrittlement	Cracking	No
				Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
630045-870-9	Spacer	Aluminum	HE	Thermal Aging	Loss of Strength	No
				General Corrosion	Loss of Material	No
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
630045-870-10	Bolt	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No
630045-870-11	Nord-Lock Washer	Stainless Steel	HE	Fatigue	Cracking	No
				Radiation Embrittlement	Cracking	No

Table 3.2-3 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
630045-870-12	Outer Port Cover	Stainless Steel	SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
				Microbiologically Influenced Corrosion	Loss of Material	No
				Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
			FE	Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
630045-877-1, -2, -3	Fuel Basket Bottom Weldment, Pads and Support Plates	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
630045-878-1, -2, -3, -4, -5, -6, -7, -8	Fuel Basket Top Weldment, Support Ring and Support Plates	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No

Table 3.2-3 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment [(3)]	Aging Mechanism	Aging Effect	Aging Management Activities
630045-881-1, -3, -4, -5, -7	Fuel Basket Fuel Tube, Cladding and Flange	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
630045-881-2, -6	Neutron Absorber	Boral	HE	Boron Depletion	Loss of Criticality Control	TLAA
				General Corrosion	Loss of Material	No
				Thermal Aging	Loss of Strength	No
				Wet Corrosion and Blistering	Change in Dimensions	No
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
				Galvanic Corrosion	Loss of Material	No
630045-881-5	Plate	Aluminum	HE	Thermal Aging	Loss of Strength	No
				Fatigue	Cracking	No
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No

Table 3.2-3 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
630045-893-1, -2, -3	Fuel Basket Support Disk	Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				General Corrosion	Loss of Material	No
				Creep	Change in Dimensions	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel (17-4 PH)	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
630045-894-1	Fuel Basket Heat Transfer Disk	Aluminum	HE	Thermal Aging	Loss of Strength	No
				General Corrosion	Loss of Material	No
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
630045-895-7, -8, -11, -12, -13, -14, -21, -22	Fuel Basket Bottom Spacers, Tie Rods and Washers	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No

Table 3.2-3 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
630045-895-10	Fuel Basket Top Nut	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
				Stress Relaxation	Loss of Preload	No
630045-902-4, -6, -7, -14, -15	Damaged Fuel Can (DFC) Wiper, Filter and Backing Screens	Stainless Steel	HE	Creep	Change in Dimensions	No
				Thermal Aging	Loss of Fracture Toughness/Loss of Ductility	No
				Radiation Embrittlement	Cracking	No
630045-902-2, -13, -16	DFC Lid Plate, Lid Bottom Plate and Dowel Pins	Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
		Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
				Creep	Change in Dimensions	No
630045-902-1, -8	DFC Collar and Upper Side Plates	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No

Table 3.2-3 Aging Management Review Results - Transportable Storage Canister (TSC) and Fuel Basket (FB) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
630045-902-5	DFC Bottom Plate	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
630045-902-9	DFC Tube Body	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
630045-902-11, -12	DFC Lift Tee and Support Ring	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
				Radiation Embrittlement	Cracking	No
				Creep	Change in Dimensions	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No

Notes:

(1) Safety functions and item/note numbers of Concrete Cask Subcomponents are identified in Table 2.5-3.
(2) Materials Legend: Steel = Carbon Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless steel (including precipitation hardened SS); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS-3); Boral = Borated aluminum-based composites; Concrete; and Spent Nuclear Fuel.
(3) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas).

Table 3.2-4 Aging Management Review Results - Vertical Concrete Cask (VCC) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent $^{(1)}$	Material $^{(2)}$	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
455-861-1	VCC Liner Shell	Steel	SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
			E-C	Radiation Embrittlement	Cracking	No
				General Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	TLAA
455-861-2, -3	Top Flange and Support Ring	Steel	SH	General Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
				Galvanic Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP

Table 3.2-4 Aging Management Review Results - Vertical Concrete Cask (VCC) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
455-861-10, -11, -12, -13, -14, -15	Base Plate Inlet Assemblies	Steel	SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
				Galvanic Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
			E-C	Pitting and Crevice Corrosion	Loss of Material	TLAA
				General Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
455-861-16, -25	Baffle Weldment and Pedestal Plate	Steel	SH	Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Galvanic Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No

Table 3.2-4 Aging Management Review Results - Vertical Concrete Cask (VCC) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent [1]	Material [2]	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
455-861-17	Nelson Stud	Steel	E-C	Pitting and Crevice Corrosion	Loss of Material	TLAA
				General Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
455-861-18, -19, -20, -21, -22, -23, -24	Outlet Vent Assemblies	Steel	SH	Galvanic Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
			E-C	Pitting and Crevice Corrosion	Loss of Material	TLAA
				General Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No

Table 3.2-4 Aging Management Review Results - Vertical Concrete Cask (VCC) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
455-862-6	Lid Bolting Hardware	Stainless Steel	SH	Stress Corrosion Cracking	Cracking	No
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Stress Relaxation	Loss of Preload	No
				Pitting and Crevice Corrosion	Loss of Material	No
				Galvanic Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
455-862-9	Baffle Weldment Cover	Stainless Steel	SH	Stress Corrosion Cracking	Cracking	No
				Radiation Embrittlement	Cracking	No
				Galvanic Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
455-863-1	VCC Lid	Steel	OD	Galvanic Corrosion	Loss of Material	External VCC Metallic Monitoring AMP
				General Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No

Table 3.2-4 Aging Management Review Results - Vertical Concrete Cask (VCC) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
455-863-1 (continued)	VCC Lid	Steel	SH	Microbiological Influenced Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
455-864-1, -2, -3, -4	Shield Plug Assembly	Steel	SH	Microbiological Influenced Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
		NSC/NSP	FE	Radiation Embrittlement	Cracking	TLAA
				Thermal Aging (NS-4-FR only)	Loss of Fracture Toughness/Loss of Ductility	TLAA
				Boron Depletion (NS-4-FR only)	Loss of Shielding Effectiveness	TLAA
455-866-1, -2, -3, -4, -5, -6, -7, -8, -9, -10, -11	Rebar	Steel	E-C	Corrosion of Reinforcing Steel	Loss of Concrete / Steel Bond	TLAA
					Loss of Material (spalling, scaling)	TLAA
					Cracking	TLAA
					Loss of Strength	TLAA

Table 3.2-4 Aging Management Review Results - Vertical Concrete Cask (VCC) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent [1]	Material [2]	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
455-866-15	Concrete Shell	Concrete	OD	Reaction with Aggregates	Cracking	Reinforced VCC Structures AMP
					Loss of Strength	Reinforced VCC Structures AMP
				Salt Scaling	Loss of Material (Spalling, Scaling)	Reinforced VCC Structures AMP
				Aggressive Chemical Attack	Cracking	Reinforced VCC Structures AMP
					Loss of Strength	Reinforced VCC Structures AMP
					Loss of Material (Spalling, Scaling)	Reinforced VCC Structures AMP
				Creep	Cracking	No
				Shrinkage	Cracking	No
				Dehydration at high temperatures	Cracking	No
					Loss of Strength	No
				Fatigue	Cracking	No
				Delayed ettringite formation	Loss of material (spalling, scaling)	No
					Cracking	No
					Loss of strength	No
				Freeze – Thaw (Above the Freeze Line)	Cracking	Reinforced VCC Structures AMP
					Loss of Material (Spalling, Scaling)	Reinforced VCC Structures AMP

Table 3.2-4 Aging Management Review Results - Vertical Concrete Cask (VCC) - YR-MPC

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
455-866-15 (continued)	Concrete Shell	Concrete	OD	Radiation Damage	Cracking	No
					Loss of Strength	No
				Leaching of Calcium Hydroxide	Loss of Strength	Reinforced VCC Structures AMP
					Increase in Porosity and Permeability	Reinforced VCC Structures AMP
					Reduction of Concrete pH (Reducing Corrosion Resistance of Steel Embedments)	Reinforced VCC Structures AMP
455-913-1, -2	Inlet Vent Supplemental Shield Assemblies	Steel	SH	Microbiological Influenced Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No

Notes:

(1) Safety functions and item/note numbers of Concrete Cask Subcomponents are identified in Table 2.5-4.
(2) Materials Legend: Steel = Carbon Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless steel (including precipitation hardened SS); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS-3); Boral = Borated aluminum-based composites; Concrete; and Spent Nuclear Fuel.
(3) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas)

Table 3.2-5 Aging Management Review Results - Vertical Concrete Cask (VCC) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
414-861-1	VCC Liner Shell	Steel	SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
			E-C	Radiation Embrittlement	Cracking	No
				General Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	TLAA
414-861-2, -3	Top Flange and Support Ring	Steel	SH	General Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Galvanic Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
414-861-16, -25	Baffle Weldment and Pedestal Plate	Steel	SH	Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Galvanic Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No

Table 3.2-5 Aging Management Review Results - Vertical Concrete Cask (VCC) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
414-861-10, -11, -12, -13, -14, -15	Base Plate and Inlet Assemblies	Steel	SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Galvanic Corrosion	Loss of Material	No
			E-C	Pitting and Crevice Corrosion	Loss of Material	TLAA
				General Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
414-861-17	Nelson Stud	Steel	E-C	General Corrosion	Loss of Material	TLAA
				Pitting and Crevice Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No

Table 3.2-5 Aging Management Review Results - Vertical Concrete Cask (VCC) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
414-861-18, -19, -20, -21, -22, -23, -24	Outlet Vent Assemblies	Steel	SH	Galvanic Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
			E-C	Pitting and Crevice Corrosion	Loss of Material	TLAA
				General Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
414-862-6	Lid Bolting Hardware	Stainless Steel	SH	Stress Corrosion Cracking	Cracking	No
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Stress Relaxation	Loss of Preload	No
				Pitting and Crevice Corrosion	Loss of Material	No
				Galvanic Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP

Table 3.2-5 Aging Management Review Results - Vertical Concrete Cask (VCC) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
414-861-27	Baffle Weldment Cover	Stainless Steel	SH	Stress Corrosion Cracking	Cracking	No
				Galvanic Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Pitting and Crevice Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
414-863-1	VCC Lid	Steel	OD	Galvanic Corrosion	Loss of Material	External VCC Metallic Monitoring AMP
				General Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
			SH	Microbiological Influenced Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No

Table 3.2-5 Aging Management Review Results - Vertical Concrete Cask (VCC) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent $^{(1)}$	Material $^{(2)}$	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
414-864-1, -2, -3, -5, -6	Shield Plug Assembly	Steel	SH	Microbiological Influenced Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
		NSC/NSP	FE	Radiation Embrittlement	Cracking	TLAA
				Thermal Aging (NS-4-FR only)	Loss of Fracture Toughness/Loss of Ductility	TLAA
				Boron Depletion (NS-4-FR only)	Loss of Shielding Effectiveness	TLAA
414-866-1, -2, -3, -4, -5, -6, -7, -8, -9, -10, -11	Rebar	Steel	E-C	Corrosion of Reinforcing Steel	Loss of Concrete / Steel Bond	TLAA
					Loss of Material (spalling, scaling)	TLAA
					Cracking	TLAA
					Loss of Strength	TLAA

Table 3.2-5 Aging Management Review Results - Vertical Concrete Cask (VCC) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent [1]	Material [2]	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
414- 866-15	Concrete Shell	Concrete	OD	Reaction with Aggregates	Cracking	Reinforced VCC Structures AMP
					Loss of Strength	Reinforced VCC Structures AMP
				Salt Scaling	Loss of Material (spalling, scaling)	Reinforced VCC Structures AMP
				Aggressive Chemical Attack	Cracking	Reinforced VCC Structures AMP
					Loss of Strength	Reinforced VCC Structures AMP
					Loss of Material (spalling, scaling)	Reinforced VCC Structures AMP
				Creep	Cracking	No
				Shrinkage	Cracking	No
				Dehydration at high temperatures	Cracking	No
					Loss of Strength	No

Table 3.2-5 Aging Management Review Results - Vertical Concrete Cask (VCC) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment	Aging Mechanism	Aging Effect	Aging Management Activities
414- 866-15 (continued)	Concrete Shell	Concrete	OD	Fatigue	Cracking	No
				Delayed ettringite formation	Loss of material (spalling, scaling)	No
					Cracking	No
					Loss of strength	No
				Freeze – Thaw (Above the Freeze Line)	Cracking	Reinforced VCC Structures AMP
					Loss of Material (spalling, scaling)	Reinforced VCC Structures AMP
				Radiation Damage	Cracking	No
					Loss of Strength	No
				Leaching of Calcium Hydroxide	Loss of Strength	Reinforced VCC Structures AMP
					Increase in Porosity and Permeability	Reinforced VCC Structures AMP
					Reduction of Concrete pH (Reducing Corrosion Resistance of Steel Embeds)	Reinforced VCC Structures AMP

Notes:

(1) Safety functions and item/note numbers of Concrete Cask Subcomponents are identified in Table 2.5-5.

(2) Materials Legend: Steel = Carbon Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless steel (including precipitation hardened SS); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS-3); Boral = Borated aluminum-based composites; Concrete; and Spent Nuclear Fuel.

(3) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas)

Table 3.2-6 Aging Management Review Results - Vertical Concrete Cask (VCC) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment [(3)]	Aging Mechanism	Aging Effect	Aging Management Activities
630045-861-1	VCC Liner Shell	Steel	SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
			E-C	Radiation Embrittlement	Cracking	No
				General Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	TLAA
630045-861-2	Top Flange	Steel	SH	General Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Galvanic Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP

Table 3.2-6 Aging Management Review Results - Vertical Concrete Cask (VCC) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
630045-861-4, -5, -6, -7	Base and Inlet Assemblies	Steel	SH	Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Galvanic Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
630045-861-8, -17	Baffle Weldment and Pedestal Plate	Steel	SH	Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Galvanic Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
630045-861-9	Nelson Stud	Steel	E-C	General Corrosion	Loss of Material	TLAA
				Pitting and Crevice Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No

Table 3.2-6 Aging Management Review Results - Vertical Concrete Cask (VCC) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
630045-861-10, -11, -12, -13, -14, -15, -16	Outlet Vent Assemblies	Steel	SH	Galvanic Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
			E-C	General Corrosion	Loss of Material	TLAA
				Pitting and Crevice Corrosion	Loss of Material	TLAA
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
630045-862-3	Lid Bolting Hardware	Stainless Steel	SH	Stress Corrosion Cracking	Cracking	No
				Radiation Embrittlement	Cracking	No
				Galvanic Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Stress Relaxation	Loss of Preload	No
				Pitting and Crevice Corrosion	Loss of Material	No
630045-861-21	Inlet Shield Bars	Steel	SH	Microbiological Influenced Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No

Table 3.2-6 Aging Management Review Results - Vertical Concrete Cask (VCC) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment [(3)]	Aging Mechanism	Aging Effect	Aging Management Activities
630045-861-22	Baffle Weldment Cover	Stainless Steel	SH	Stress Corrosion Cracking	Cracking	No
				Radiation Embrittlement	Cracking	No
				Galvanic Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
630045-863-1, -2, -3, -4, -5, -6	VCC Lid Assembly	Steel	OD	Galvanic Corrosion	Loss of Material	External VCC Metallic Monitoring AMP
				General Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
		Steel	SH	Microbiological Influenced Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
				Radiation Embrittlement	Cracking	No
		Concrete	FE	Radiation Embrittlement	Cracking	No

Table 3.2-6 Aging Management Review Results - Vertical Concrete Cask (VCC) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
630045-866-1, -2, -4, -5, -6, -7, -8, -9, -10, -11, -26, -27	Rebar	Steel	E-C	Corrosion of Reinforcing Steel	Loss of Concrete/Steel Bond	TLAA
					Loss of Material (spalling, scaling)	TLAA
					Cracking	TLAA
					Loss of Strength	TLAA
630045-866-15	Concrete Shell	Concrete	OD	Reaction with Aggregates	Cracking	Reinforced VCC Structures AMP
					Loss of Strength	Reinforced VCC Structures AMP
				Salt Scaling	Loss of Material (Spalling, Scaling)	Reinforced VCC Structures AMP
				Aggressive Chemical Attack	Cracking	Reinforced VCC Structures AMP
					Loss of Strength	Reinforced VCC Structures AMP
					Loss of Material (Spalling, Scaling)	Reinforced VCC Structures AMP
				Creep	Cracking	No
				Shrinkage	Cracking	No
				Dehydration at high temperatures	Cracking	No
					Loss of Strength	No
				Fatigue	Cracking	No

Table 3.2-6 Aging Management Review Results - Vertical Concrete Cask (VCC) - MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent [1]	Material [2]	Storage Operation Environment [3]	Aging Mechanism	Aging Effect	Aging Management Activities
630045-866-15 (continued)	Concrete Shell	Concrete	OD	Delayed ettringite formation	Loss of material (spalling, scaling)	No
					Cracking	No
					Loss of strength	No
				Freeze – Thaw (Above the Freeze Line)	Cracking	Reinforced VCC Structures AMP
					Loss of Material (Spalling, Scaling)	Reinforced VCC Structures AMP
				Radiation Damage	Cracking	No
					Loss of Strength	No
				Leaching of Calcium Hydroxide	Loss of Strength	Reinforced VCC Structures AMP
					Increase in Porosity and Permeability	Reinforced VCC Structures AMP
					Reduction of Concrete pH (Reducing Corrosion Resistance of Steel Embedments)	Reinforced VCC Structures AMP

Notes:

(1) Safety functions and item/note numbers of Concrete Cask Subcomponents are identified in Table 2.5-6.

(2) Materials Legend: Steel = Carbon Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless steel (including precipitation hardened SS); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS-3); Boral = Borated aluminum-based composites; Concrete; and Spent Nuclear Fuel.

(3) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas)

Table 3.2-7 Aging Management Review Results - Transfer Cask (TFR) - YR-MPC / MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material[2)]	Storage Operation Environment [(3)]	Aging Mechanism	Aging Effect	Aging Management Activities
455-860-1	Bottom Plate	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
455-860- 2	Inner Shell	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
455-860-4	Outer Shell	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Galvanic Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
455-860-5	Trunnion	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Wear	Loss of Material	Transfer Cask AMP

Table 3.2-7 Aging Management Review Results - Transfer Cask (TFR) - YR-MPC / MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
455-860-8	Neutron Shield	NSP	FE	Radiation Embrittlement	Cracking	TLAA
				Thermal Aging	Loss of Fracture Toughness	TLAA
				Boron Depletion	Loss of Shielding Effectiveness	TLAA
455-860-9	Top Plate	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
455-860-10	Door Rail (5)	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Galvanic Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Wear	Loss of Material	Transfer Cask AMP
455-860-13, -29, and 455-918-5	Door Lock Bolt/Stop	Stainless Steel	OD	Pitting and Crevice Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Stress Corrosion Cracking	Cracking	No
				Stress Relaxation	Loss of Preload	No

Table 3.2-7 Aging Management Review Results - Transfer Cask (TFR) - YR-MPC / MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
455-860-14	Retaining Ring	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Galvanic Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
455-860 -11, -12	Shield Door Assembly A and B	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Wear	Loss of Material	Transfer Cask AMP
455-860-3	Gamma Shield Brick	Lead	FE	None Identified	None Identified	No
455-860-15, -34	Retaining Ring & Strut Bracket Bolt	Stainless Steel (Ferritic)	OD	Pitting and Crevice Corrosion	Loss of Material	No
				Galvanic Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Stress Relaxation	Loss of Preload	No
				Microbiological Influenced Corrosion	Loss of Material	No
455-860-17	Connector	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Wear	Loss of Material	Transfer Cask AMP

Table 3.2-7 Aging Management Review Results - Transfer Cask (TFR) - YR-MPC / MPC-LACBWR

Applicable License Drawing/Item No.	Subcomponent (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
455-860-33	Strut Bracket	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Galvanic Corrosion	Loss of Material	Transfer Cask AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Wear	Loss of Material	Transfer Cask AMP
455-859 -1, -2, -3, -4, -5	Transfer Adapter Assembly	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Wear	Loss of Material	Transfer Cask AMP

Notes:

(1) Safety functions and item/note numbers of YR-MPC/MPC-LACBWR Transfer Cask Subcomponents are identified in Table 2.5-7.
(2) Materials Legend: Steel = Carbon Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless steel (including precipitation hardened SS); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS3); Lead; Boral = Borated aluminum-based composites (Boral); Concrete; and SNF = Spent Nuclear Fuel
(3) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas).
(4) Component coatings and operational conditions inspected and maintained under the TFR Maintenance Program.
(5) Sliding surfaces of the TFR shield doors and rail components are lubricated with spent fuel pool compatible lubricant such as Neolube or equivalent.

Table 3.2-8 Aging Management Review Results - Transfer Cask (TFR) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material[2)]	Storage Operation Environment [(3)]	Aging Mechanism	Aging Effect	Aging Management Activities
414-860-1	Bottom Plate	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
414-860- 2	Inner Shell [(4)]	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
414-860-4	Outer Shell	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Galvanic Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No

Table 3.2-8 Aging Management Review Results - Transfer Cask (TFR) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
414-860-5	Trunnion	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Wear	Loss of Material	Transfer Cask AMP
414-860-8	Neutron Shield	NSP	FE	Radiation Embrittlement	Cracking	TLAA
				Thermal Aging	Loss of Fracture Toughness	TLAA
				Boron Depletion	Loss of Shielding Effectiveness	TLAA
414-860-9	Top Plate	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
414-860-10	Door Rail (5)	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Galvanic Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Wear	Loss of Material	Transfer Cask AMP

Table 3.2-8 Aging Management Review Results - Transfer Cask (TFR) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
414-860-13 and 414-917-5	Door Lock Bolt/Stop	Stainless Steel	OD	Pitting and Crevice Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Stress Corrosion Cracking	Cracking	No
				Stress Relaxation	Loss of Preload	No
414-860-14	Retaining Ring	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Galvanic Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
414-860 -11, -12	Shield Door Assembly A and B	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Wear	Loss of Material	Transfer Cask AMP
414-860-3	Gamma Shield Brick	Lead	FE	None Identified	None Identified	No
414-860-15	Retaining Ring Bolt	Stainless Steel (Ferritic)	OD	Pitting and Crevice Corrosion	Loss of Material	No
				Galvanic Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Stress Relaxation	Loss of Preload	No
				Microbiological Influenced Corrosion	Loss of Material	No

Table 3.2-8 Aging Management Review Results - Transfer Cask (TFR) - CY-MPC

Applicable License Drawing/Item No.	Subcomponent (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
414-860-17	Connector	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Wear	Loss of Material	Transfer Cask AMP
455-859 -1, -2, -3, -4, -5	Transfer Adapter Assembly	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
				Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
				Wear	Loss of Material	Transfer Cask AMP

Notes:

(1) Safety functions and item/note numbers of CY-MPC Transfer Cask Subcomponents are identified in Table 2.5-8.
(2) Materials Legend: Steel = Carbon Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless steel (including precipitation hardened SS); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS3); Lead; Boral = Borated aluminum-based composites (Boral); Concrete; and SNF = Spent Nuclear Fuel
(3) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas).
(4) Component coatings and operational conditions inspected and maintained under the TFR Maintenance Program.
(5) Sliding surfaces of the TFR shield doors and rail components are lubricated with spent fuel pool compatible lubricant such as Neolube or equivalent.

Table 3.2-9 NAC-MPC Spent Fuel Assemblies Aging Management Review (AMR) Results

Structure, System, or Component	Intended Safety Function [(1)]	Material [2)]	Storage Operation Environment [(3)]	Aging Mechanism	Aging Effect	Aging Management Activities
Fuel rod cladding	CO, CR, RE, SH, SR, TH	Zirconium-based alloy (Zircaloy-2, Zircaloy-4, ZIRLO™, or M5®)	HE	Oxidation	Loss of Load Bearing Capacity	No
				Pitting Corrosion	Loss of Material	No
				Galvanic Corrosion	Loss of Material	No
				Stress Corrosion Cracking	Cracking	No
				Hydride-Induced Embrittlement	Loss of Ductility	No
				Delayed Hydride Cracking	Cracking	No
				Low-Temperature Creep	Changes in Dimensions	No
				Radiation Embrittlement	Loss of Strength	No
				Fatigue	Cracking	No
				Mechanical Overload	Cracking	No
Fuel rod cladding	CO, CR, RE, SH, SR, TH	Stainless Steel Alloy	HE	General Corrosion	Loss of Material	No
				Stress Corrosion Cracking	Cracking	No
				Pitting Corrosion	Loss of Material	No
				Stress Rupture	Cracking	No
				Strain Rate Embrittlement	Cracking	No
				Hydrogen-Induced Degradation	Cracking	No
				Helium Embrittlement	Cracking	No
				Fatigue	Cracking	No
				Fission Product Cladding Interaction	Cracking	No

Table 3.2-9 NAC-MPC Spent Fuel Assemblies Aging Management Review (AMR) Results

Structure, System, or Component	Intended Safety Function (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
Guide tubes (PWR) or water channels (BWR)	RE, SR	Zirconium-based alloy or stainless steel	HE	Creep	Change in Dimensions	No
				Hydriding	Change in Dimensions	No
				Radiation Embrittlement	Loss of Strength	No
				Fatigue	Cracking	No
Spacer grids	CR, RE, SR, TH	Zirconium-based alloy or stainless steel	HE	Creep	Change in Dimensions	No
				Hydriding	Change in Dimensions	No
				Radiation Embrittlement	Loss of Strength	No
				Fatigue	Cracking	No
		Inconel	HE	Creep	Change in Dimensions	No
				General Corrosion	Loss of Material	No
				Stress Corrosion Cracking	Cracking	No
				Radiation Embrittlement	Loss of Strength	No
				Fatigue	Cracking	No

Table 3.2-9 NAC-MPC Spent Fuel Assemblies Aging Management Review (AMR) Results

Structure, System, or Component	Intended Safety Function (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
Lower and Upper End Fittings	CR, RE, SR	Stainless steel	HE	Creep	Change in Dimensions	No
				General Corrosion	Loss of material	No
				Stress Corrosion cracking	Cracking	No
				Radiation Embrittlement	Loss of Strength	No
				Fatigue	Cracking	No
		Inconel	HE	Creep	Change in Dimensions	No
				General Corrosion	Loss of Material	No
				Stress Corrosion Cracking	Cracking	No
				Radiation Embrittlement	Loss of Strength	No
				Fatigue	Cracking	No
Fuel channel (BWR)	CR, TH	Zirconium-based alloy	HE	Creep	Change in Dimensions	No
				Hydriding	Change in Dimensions	No
				Radiation Embrittlement	Loss of Strength	No
				Fatigue	Cracking	No

Table 3.2-9 NAC-MPC Spent Fuel Assemblies Aging Management Review (AMR) Results

Structure, System, or Component	Intended Safety Function (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities
Poison rod assemblies (PWR)	CR	Stainless steel	HE	Creep	Change in Dimensions	No
				General Corrosion	Loss of Material	No
				Stress Corrosion Cracking	Cracking	No
				Radiation Embrittlement	Loss of Strength	No
				Fatigue	Cracking	No

Notes:

(1) Safety functions of PWR and BWR SNF Subcomponents are identified in Tables 2.5-9.

(2) Materials Legend: Steel = Carbon Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); SS = Stainless steel (including precipitation hardened SS); AL= Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS3); BAL = Borated aluminum-based composites (Boral); and C = Concrete.

(3) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas).

3.3 TIME-LIMITED AGING ANALYSES (TLAAs)

This section lists and describes the proposed TLAAs identified as required in Section 3.2 for the NAC-MPC System SSCs. The TLAAs will incorporate current design basis analyses and expand as required to document the performance of the identified SSC subcomponents for the intended 60-year component performance including the planned 40-year period of extended operation. The completed TLAAs are provided in Appendix B.

In-scope SSC that are subject to a potential aging effect are addressed either through Time-Limited Aging Analysis (TLAA) or by an Aging Management Program (AMP). TLAAs that can adequately predict degradation associated with identified aging effects and can be reconfirmed for the period of extended operation, do not require additional Aging Management Activities (AMAs). This section discusses the criteria used to identify TLAAs and the evaluation and disposition of the identified TLAAs for the extended period of operation. In accordance with 10 CFR 72.240(c)(2), the TLAAs demonstrate that SSC ITS will continue to perform their intended safety function for the period of extended operation.

3.3.1 TLAA Identification Criteria

The following criteria defined in NUREG-1927 [3.9.2] are used to identify TLAAs for existing SSC with a time dependent operating life:

(1) Involve SSCs important to safety within the scope of the renewal
(2) Consider the effects of aging,
(3) Involve time limited assumptions (e.g., 20-year) that are explicit in the analysis,
(4) Determined to be relevant in making a safety determination,
(5) Provides conclusions, or the basis for conclusions, regarding the capability of the SSC to perform its intended safety function through the operating term, and
(6) Are contained or incorporated by reference in the design bases.

3.3.2 TLAA Identification Process and Results

Design documents for the NAC-MPC System were reviewed against the TLAA identification criteria discussed in Section 3.3.1. These included the CoC, NRC Safety Evaluation Reports (SERs), and Technical Specifications for the NAC-MPC System, NAC-MPC System FSAR, docketed licensing correspondence, and generic calculations and site-specific calculations and evaluations as defined in Section 3.8.

The proposed TLAAs are identified in the AMR Tables 3.2-1 through 3.2-9 for the NAC-MPC System TSCs and Fuel Baskets, VCCs, and Transfer Casks.

3.3.3 Evaluation and Disposition of Identified TLAAs

3.3.3.1 Fatigue Evaluation of NAC-MPC System Components for Extended Storage (NAC Calculation No. 30013-2001), Revision 0

The potential fatigue of the NAC-MPC SSCs (e.g., canisters and fuel baskets for YR-MPC, CY-MPC, and MPC-LACBWR systems) were evaluated in a TLAA for service conditions over the extended period of operation. The NAC-MPC canisters satisfy all conditions stipulated in NB-3222.4(d)(1) through (6), and the fuel baskets satisfy all conditions stipulated in NG-3222.4(d)(1) through (4) for a 60-year service life. Therefore, the NAC-MPC canisters and fuel baskets do not require fatigue analysis for cyclic service for 60-years of extended storage conditions.

3.3.3.2 Corrosion Analysis of NAC-MPC Steel Components for Extended Storage Operation (NAC Calculation No. 30013-2003, Revision 2)

The TLAA evaluated the general corrosion of NAC-MPC Vertical Concrete Cask (VCC) carbon steel components at a constant rate of 0.003 inch per year over the entire 60-year period of extended operation resulting in a total corrosion allowance of 0.18 inch. The total corrosion allowance is evaluated for the different VCC steel components and it is determined not to have an adverse effect on the ability of the VCC assembly to perform its intended structural, thermal and shielding functions. Also, there are no credible aging mechanisms that would affect the VCC steel internals to result in significant pitting or crevice corrosion. Therefore, pitting and crevice corrosion will have no adverse effects on the ability of the VCC assembly to perform its intended safety functions.

The structural evaluation of the VCC for the bottom lift by hydraulic jacks shows that the maximum bearing stress in the concrete and the maximum stresses in the pedestal with corrosion after a 60-year service life remain within the allowable stress limits. In addition, the 0.18-inch corrosion allowance on the opposite side of the plates to which the nelson studs are welded will not adversely impact the design function of the Nelson studs. Finite element analyses of the VCC pedestals with the maximum corrosion at the end of the 60-year service period show that the maximum stress intensities in the base and ring remain well below the allowable stress limits. The margins of safety in the base and ring for the bottom lift with hydraulic jacks, with the maximum corrosion at the end of the 60-year service life, are > 10 and 3.05, respectively.

The structural evaluation of the NAC-MPC VCC for dead load, live load, flood, tornado wind, and seismic loading did not take any structural credit for the VCC steel liner, and therefore, it is concluded that any reduction in the VCC liner thickness resulting from corrosion does not change the results of the VCC analysis for these load conditions.

The structural evaluation for thermal loading concludes that a reduction of the VCC steel liner thickness due to corrosion would result in a negligible change in the thermal stresses in the concrete and rebar. For the steel liner, the thermal stress is reduced due to corrosion since the reduction of the liner thickness will result in a smaller through-wall thermal gradient. Note that this reduction of thermal gradient is significantly overshadowed by the reduction of the thermal gradient due to decay of the canister heat loads over the 60-year extended service period.

The analysis of local damage to the NAC-MPC VCC concrete shell due to tornado missile impacts did not take any structural credit for the VCC steel liner, and therefore, it is concluded that any reduction in the VCC liner thickness resulting from corrosion does not change the results of the VCC analysis for tornado missile impact. The structural evaluation of the VCC assembly for strength required to prevent perforation by the design-basis armor piercing tornado generated missile shows that the corroded lid thickness of 1.14 inches after 60 years remains sufficient to prevent missile perforation.

The structural evaluation of the NAC-MPC VCC assembly for the VCC 6-inch drop includes an evaluation of the concrete shell and the pedestal. The evaluation of the concrete shell did not take any structural credit for the VCC steel liner, and therefore, it is concluded that any reduction in the VCC liner thickness resulting from corrosion does not change the results of the VCC concrete shell for this load conditions. The evaluation of the pedestal concluded that the maximum deformation of the pedestal due to the drop will increase to 0.69-inch, resulting in a 14% reduction of the air inlet cross-section area, which is bounded by the half inlets blocked condition. Furthermore, it is concluded that the weldment plate (and canister) will not “bottom-out” and, therefore, the canister acceleration loads will be lower than those for calculated based on the nominal plate thicknesses.

The structural evaluation of the NAC-MPC VCC assembly for the tip-over concluded that general corrosion of the steel inner shell will reduce the overall beam-bending and ring-bending stiffness of the VCC, which will slightly reduce the acceleration loads that are imparted to the canister and basket components.

The thermal analysis concludes that corrosion of the steel plates that line the VCC air passage will improve the surface properties with respect to thermal performance, but the expansion of the rust layer into the air passage could reduce the air flow cross section by up to 10%. The net effect of the corrosion of the steel surfaces that line the air passage on the thermal performance of the system is insignificant.

The NAC-MPC VCC shielding analysis concludes that the reduction in gamma shielding resulting from loss of steel due to corrosion over the extended storage period is more than offset by the decay of the source over the same timeframe.

Additionally, it has been determined that the potential impact of pitting corrosion and crevice corrosion on the performance of VCC sheltered components would not have a deleterious effect on the functional or safety performance of the VCC liner, shield plug, VCC lid, baseplate/pedestal, inlet/outlet vents, Nelson studs, rebar and lift lug threaded rebar. The size and thickness of the carbon steel of these components and the limited number of crevices in the construction of the VCC would limit any effects of pitting and crevice corrosion on the shielding or thermal performance of the VCC carbon steel components. In addition, as both pitting and crevice corrosion would generally appear with general corrosion of the surfaces, the identification of these separate corrosion modalities would be limited by the inability to observe the surfaces below the areas of general corrosion. In conclusion, the TLAA has evaluated the ability of the sheltered VCC carbon steel components to withstand the potential uniform loss of up to 0.18 inches in thickness and still maintain the FSAR analyzed structural, thermal and shielding performance requirements and safety functions. In addition, steel components fully encased in concrete (i.e., rebar, Nelson studs) form a thin oxide layer (passive film) due to the alkaline environment of the concrete which reduces the rate of corrosion to 0.1 μm/year. Similarly, the embedded surfaces of partially embedded components (i.e., inner liner, top flange, bottom plate, inlet and outlet side and top plates, stand and outer plates) will be negligible compared to the exposed surfaces of these components. This extent of corrosion would not affect the performance of the embedded and partially embedded components and factors of safety are greater than 3 compared to yield strength and greater than 5 compared to ultimate strength throughout the 60-year period of extended service.

Exterior steel surfaces which are accessible (e.g., VCC lid exterior, and exposed top flange surfaces), fully exposed to the environment, and coated with primer and paint are assumed to be protected from corrosion for the 60-year period of extended operation. If minor defects in coating are identified during performance of External VCC Metallic Components AMP, the area of coating loss will be cleaned and recoated to maintain the condition of the external metallic components and maintain their intended safety functions. The low rate of corrosion of these carbon steel components between AMP inspections and corrective actions by Licensees ensure these components maintain their required factors of safety. Therefore, stresses and factors of safety for the exposed portion of the lift anchor/lift lugs, and other coated exterior steel components remain unchanged.

3.3.3.3 Aging Analysis for NAC-MPC Neutron Absorber and Neutron Shield Components (Storage/Transfer) (NAC Calculation No. 30013-5001), Revision 0

NAC-MPC system was evaluated for:

- Depletion of the neutron absorber boron-10 content in the basket
  - Considering the extremely conservative assumption of all neutrons emitted by the design basis fuel being absorbed in the neutron absorber sheets, the service life is well over 60 years.
  - A bounding depletion fraction was estimated at $1\text{x}10^{-9}$ per year. At 60 years < 1% of the B-10 in the absorber sheets will be depleted.
  - There is no impact on the criticality safety of the system from such a small depletion percentage (only 75% of the minimum B-10 content is credited in the criticality analysis).
  - In a dry storage system, the neutron flux is primarily composed of non-thermal neutrons which will not deplete the neutron absorber (B-10 has primarily a thermal neutron absorption cross section).

- Depletion of the neutron absorber boron-10 in the NAC-MPC system radiation shield components
    - Considering the fluxes produced by design basis neutron sources emitted by the design basis fuel assembly, the service life in the context of boron depletion of all neutron shield components in the VCC and transfer cask is well over 60 years.
    - At 60 years < 1% of the B-10 in the neutron shield will be depleted in the most limiting neutron shield component (MPC transfer cask bottom/door transfer).
- Radiation embrittlement in the cask radiation shield components
    - Embrittlement is not a concern for the cask neutron shield components as they are captured within shells and do not perform a structural function.
    - Total gamma and neutron fluxes will not significantly impact system performance over a 60-year design life.

3.3.3.4 Evaluation of Stainless-Steel Clad Fuel for Fatigue in Storage (NAC Calculation No. 30013-3004, Revision 0)

The storage of stainless-steel clan spent fuel for fatigue was evaluated for the NAC-MPC System. All the NAC-MPC Systems in current operation are storing stainless steel clad spent nuclear fuel including CY-MPC. YR-MPC and LACBWR-MPC systems.

The purpose of the analysis was to determine what effect storage operations in the NAC-MPC System would have on the fatigue of the stainless steel clad, the potential for fuel degradation due to long term storage for a period of 60 years. In the analysis, stresses developed in the stainless-steel fuel cladding when in dry storage conditions are calculated from thermal cycles developed by daily and seasonal ambient air temperature fluctuations and cooling from temperature at maximum heat load to ambient temperature. The induced cyclic stresses in the cladding are evaluated for fatigue over a time span of 60 years. The analysis considers the Connecticut Yankee, Yankee Rowe, and LACBWR fuel.

The Connecticut Yankee, Yankee Rowe, and LACBWR fuel with stainless steel cladding are evaluated for fatigue from stresses developed from ambient air temperature fluctuations when in dry storage. The analysis is based on the fatigue assessment provided in NUREG-2214 for zirconium-based cladding. Daily and seasonal temperature variations and their associated thermal stresses are considered. The number of thermal cycles is based on a time span of 60 years. The evaluation also includes the effect of the fuel cooling from temperature at maximum heat load to ambient.

The cumulative damage assessment from the different air temperature fluctuations and their associated thermal stresses shows a large margin against fatigue failure in the cladding for all three fuels. Calculated alternating stresses from the thermal cycles are very similar for the different fuel geometries and the initial internal pressure considered. The alternating stress induced by daily temperature fluctuations is below the material endurance limit by almost a factor of two, and this predicts that the daily alternating stress cycles will not cause fatigue failure over an infinite time span. The seasonal thermal cycles are the largest contributor to the cumulative fatigue damage. The fatigue damage from the fuel cooling from maximum temperature to ambient is approximately one-quarter of that from the seasonal fluctuations.

3.4 AGING MANAGEMENT PROGRAMS (AMPs)

This section lists and describes the proposed AMPs identified as required in Section 3.2 for the NAC-MPC System SSCs. The AMPs are based on the current NRC guidance in NUREG-1927, Revision 1 [3.9.2] and NRC guidance in NUREG-2214, Managing Aging Processes in Storage (MAPS) Report [3.9.4], and other recently re-certified dry storage systems. The in-scope SSC's that are subject to aging effects that require either an AMP or TLAA are identified in Section 3.2. Section 3.3 discusses the TLAAs used to evaluate aging effects and associated aging mechanism(s) and demonstrate that they do not adversely affect the ability of the SSC to perform their intended functions during the extended storage period. Those aging effects that are not adequately addressed by TLAA require an AMP. The AMP elements used to manage aging effects in the in-scope SSC are discussed in this section.

3.4.1 Aging Effects Subject to Aging Management

Aging effects that could result in loss of in-scope SSC intended functions are required to be managed during the extended storage period. The aging effects that require management are discussed in Section 3.2 and are summarized in AMR Tables 3.2-1 through 3.2-9 for the NAC-MPC System TSCs and Fuel Baskets, VCCs, Transfer Casks and Transfer Adapters, and SNF Assemblies. Many aging effects are dispositioned for the extended storage period using TLAA, as discussed in Section 3.3. An AMP is used to manage those aging effects that are not dispositioned by TLAA. The AMP is described in Section 3.4.2.

3.4.2 Aging Management Program Description

The AMP that manages each of the identified aging effects for all in-scope SSC is described in this section. The AMP consists of the existing surveillance requirements in the NAC-MPC System Technical Specifications, with additional examinations to address aging that could potentially occur during the period of extended operation.

The identified AMPs are as follows:

- AMP-1 - Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSCs) (Table A-1)
- AMP-2 - Aging Management Program for Internal Vertical Concrete Casks (VCC) Metallic Components Monitoring (Table A-2)
- AMP-3 - Aging Management Program for External Vertical Concrete Casks (VCC) Metallic Components Monitoring (Table A-3)
- AMP-4 - Aging Management Program for NAC Reinforced Vertical Concrete Cask (VCC) Structures – Concrete Monitoring (Table A-4)
- AMP-5 - Aging Management Program for Transfer Casks (TFR) and Transfer Adapters (Table A-5)

The proposed AMPs are presented in Appendix A of this application.

3.4.3 Aging Management Program Deviations from MAPS Final Report (NUREG-2214)

3.4.3.1 AMP-1 Aging Management Program for Localized Corrosion Cracking and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSCs)

In lieu of utilizing the proposed inspection guidelines and acceptance criteria proposed in Table 6.2 of NUREG-2214, NAC-MPC users intend to utilize the inspections guidelines and acceptance criteria provided in EPRI Report TR-3002008193 [3.9.16] and ASME Code Case N-860 (3.9.368) for supplemental examination of major indications as documented in the proposed AMP. To support identification of the most susceptible TSCs to SSC, all NAC-MPC user ISFSIs and loaded TSCs were evaluated and ranked utilizing EPRI Report TR-3002005371 [3.9.15].

For examination of TSC welds and heat affected zones (HAZs) qualified VT-3 inspection methods will be utilized. Certain inspection results require a supplemental examination per ASME Code Case N-860 (3.9.368). If a supplemental examination is not possible or is unable to provide sufficient data, an analysis shall be performed as provided for by Section-2440. TSC surfaces outside of the welds and HAZs, a direct or remote general visual inspection will be conducted. If issues are identified during the general visual inspection of non-welded TSC surfaces, supplemental examinations can be performed with VT-3 and VT-1 equipment and methods.”

3.4.3.2 AMP-2 Aging Management Program for Internal Vertical Concrete Casks (VCC) Metallic Components Monitoring

The VCC internal metallic components have been evaluated by TLAA Corrosion Analysis of NAC-MPC Steel Components for Extended Storage Operation (NAC Calculation No. 30013-2003) to not require inspection for general corrosion, pitting or crevice corrosion. The proposed AMP covers the opportunistic inspection of VCC internals during performance of TSC inspections per AMP No. 1. A general visual inspection using direct and remote methods will be performed on the VCC internals during performance of the TSC inspections per AMP No. 1 in lieu of performing a VT-3 inspection. A separate AMP has been proposed for the external inspections of VCC metallic components which are performed in concert with AMP-4 for Reinforced Vertical Concrete Cask.

3.4.3.3 AMP-3 Aging Management Program for External Vertical Concrete Casks (VCC) Metallic Components Monitoring

The VCC external metallic components have been evaluated by TLAA Corrosion Analysis of NAC-MPC Steel Components for Extended Storage Operation (NAC Calculation No. 30013-2003) to not require inspection for general corrosion, pitting or crevice corrosion as long as the components coating is maintained for the PEO. The proposed AMP requires a general condition assessment for coating damage of external metallic VCC components by direct observation. If minor areas of coating deterioration are identified during the inspection, corrective actions can be implemented to clean and recoat the affected components. If major coating deterioration of external VCC components is identified, Condition Reports per the Corrective Action Program will document the condition and develop an appropriate response to it.

3.4.3.4 AMP-4 Aging Management Program for NAC Reinforced Vertical Concrete Cask Structures – Concrete Monitoring

A general visual inspection of accessible external concrete surfaces will be performed utilizing the ACI 349.3R-02 [3.9.155] Tier concrete evaluation criteria. VCC fully embedded components including Nelson studs, and rebar were evaluated by TLAA Corrosion Analysis of NAC-MPC Steel Components for Extended Storage Operation (NAC Calculation No. 30013-2003) to not require inspection for general corrosion, pitting or crevice corrosion for the PEO. Similarly, the rate of corrosion on the embedded surfaces of is negligible as compared to the exposed surfaces of these components. Therefore, for the partially embedded components (i.e., inner liner, top flange, bottom plate, inlet and outlet top and side plates, stand and outer plates) there is no change to the factors of safety for these components over the PEO.

Based on the NRC evaluations performed on the NAC-MPC System shielding performance [3.9.76], it has been shown that the ACI 349.3R-02 Tier 2 concrete evaluation criteria are sufficient to ensure that the VCC concrete structure has not deteriorated and that the performance the proposed periodic shielding tests/evaluations was not required. It was noted in the NRC evaluations [3.9.76] that the shielding analysis for the MPC-LACBWR (or similar contents) would require additional analysis beyond the original design basis contents analysis. However, the LACBWR fuel is low burnup (maximum burnup of 21,532), will have been cooled for over 48 years (last discharge date of 1987) at the first required examination in the PEO in 2032, and all five MPC-LACBWR VCCs have heat loads of ≤ 2.2 kW (at the current time). Therefore, NAC is confident that the MPC-LACBWR VCC dose rates will be below the design basis limits established in the FSAR for the CY-MPC VCC design, and therefore the MPC-LACBWR VCCs also do not require periodic shielding inspections. It is noted that all NAC-MPC ISFSIs will continue to be monitored for compliance with 10 CFR 72.104.

3.4.3.5 AMP-5 Aging Management Program for Transfer Casks (TFR) and Transfer Adapters

A general visual inspection of the internal and external surfaces of the TFRs and Transfer Adapters are performed every five years when the equipment has been in service, or within one year of next use. In addition, the accessible trunnion surfaces are dye penetrant (PT) examined for the presence of fatigue cracks in accordance with ASME Code, Section III, Subsection NF, NF-5350.

3.5 PERIODIC TOLLGATE ASSESSMENTS

Periodic tollgate assessments (e.g., learning aging management per NUREG-1927 [3.9.2]) and as described in NEI 14-03 [3.9.3] are an important part of a learning, operations-based aging management program. General Licensees (GLs) are required to perform and document periodic tollgate assessments on the state of knowledge of aging-related operational experience, research, monitoring, and inspections to ascertain the ability of in-scope NAC-MPC System SSCs to continue to perform their intended safety functions throughout the renewed period of extended operation. This section of the CoC renewal application described the general requirements for the periodic tollgate assessments that must be addressed in the programs and procedures that are established, maintained, and implemented by each GL for the AMPs.

Each GL shall complete the initial tollgate assessment within 5-years following the 20th in-service year of the first cask loaded at each site or 6-years after the effective date of the CoC renewal, whichever is later. Subsequent tollgate assessments will be performed at a 10-year (± 1 year) frequency thereafter. The initial tollgate assessment is timed to allow the initial round of AMP inspections to be completed at the GL's site before the initial tollgate assessment, such that the Operating Experience (OE) gained from the initial round of AMP inspections can be evaluated and assessed. The 10-year frequency for subsequent tollgate assessments reflects the risk significance of the aging effects managed by AMPs. However, if the results of previous tollgate assessments indicate unanticipated or accelerated aging effects, the period for follow-on assessments will be reduced based upon the timing of the aging mechanisms identified and their risk significance. The basis of any adjustments in the tollgate assessment frequency shall be included in the tollgate assessment report.

At a minimum, the periodic tollgate assessments to be performed by each GL shall consider the OE related to the aging effects managed by the AMPs from the GL's completed inspections and those of other GLs that use the NAC-MPC System. The assessments will also consider new information on relevant aging effects from related industry OE, research findings, monitoring data and inspection results, NRC generic communications, DOE research updates, AMID data base, and relevant information/reports from industry organizations such as NEI, EPRI, and INPO, as applicable. The evaluation of aggregated OE will be performed to identify any new aging effects or aging mechanisms that may be applicable to the in-scope SSCs of the NAC-MPC System or are not adequately managed by the current AMPs and/or TLAAs. The assessment will also evaluate if continued safe storage is expected until the next tollgate assessment, or if additional aging management activities are required to address newly identified aging effects requiring management.

Tollgate assessment finding that require corrective actions shall be documented and evaluated in accordance with the GL's corrective action program. Proposed changes to the AMPs and/or TLAAs described in the FSAR to address newly identified aging effects shall be evaluated in accordance with 10 CFR 72.48 to determine if the proposed changes require prior NRC approval prior to implementation.

Each GL shall document the periodic tollgate assessment in a report, which will document the following information, at a minimum:

- The sources of OE, aggregated research findings, monitoring date, and inspection results considered in the assessment;
- Summary of the research findings, OE, monitoring data and inspection results;
- Potential impact, if any, of the research findings, OE, monitoring data, and inspection results on the AMPs and/or TLAAs for the in-scope SSCs;
- Recommended corrective actions to be implemented to address newly identified aging effects that are not adequately managed by the existing AMPs and/or TLAAs; and
- Summary and conclusions.

The tollgate assessment report(s) will be maintained by the GL as a permanent record in accordance with the requirements of their QA program and will be available for NRC inspection. A copy of each tollgate assessment report will also be provided to the Certificate Holder (CH) NAC International. As deemed appropriate, the tollgate assessment reports will be disseminated through an industry organization (e.g., NEI, EPRI, INPO).

3.6 FUEL RETREIVABILITY

Retrievability is the ability to readily retrieve spent nuclear fuel from storage for further processing and disposal in accordance with 10 CFR 72.122 (l). ISG-2, Revision 2 [3.9.18] provides staff guidance on the subject of ready retrieval as "the ability to safely remove the spent fuel from storage for further processing or disposal. Per ISG-2, the NRC interprets this regulation that a storage system be designed to allow ready retrieval in the initial design, amendments to the design, and in license renewal, through the aging management of the design.

In order to demonstrate the ability for ready retrieval, a licensee should demonstrate it has the ability to perform any of the three options listed below. These options may be utilized individually or in any combination or sequence, as appropriate.

A. Remove individual or canned spent fuel assemblies from wet or dry storage,
B. Remove a canister loaded with spent fuel assemblies from a storage cask/overpack,
C. Remove a cask loaded with spent fuel assemblies from the storage location.

The NAC-MPC storage system is designed to allow ready retrieval of the SNF assemblies for further processing and disposal, in accordance with 10 CFR 72.122(l) by either option A. or option B above. Under Option A, the NAC-MPC canisters are designed for opening of the canister at a suitable facility for removal and transfer of the individual or canned spent fuel assemblies, and under Option B by transfer of a loaded NAC-MPC canister to the approved and NRC certified NAC-STC transport cask system (CoC No. 71-9235) [3.9.152] for transport off-site without the need for repackaging.

The results of the AMR show there are no credible aging effects in the SNF assemblies that require management during the period of extended storage. Low burnup (≤ 45 GWd/MTU), intact and damaged (loaded in damaged fuel cans [DFCs]), zircaloy and stainless steel clad PWR and BWR SNF assemblies are stored in the NAC-MPC storage system. No high burnup fuel assemblies are stored in any of the deployed NAC-MPC storage systems. Degradation of the cladding of the low burnup fuel will not occur during the period of extended operation because the inert helium atmosphere inside the canister is maintained. Corrosion and chloride-induced stress corrosion cracking (CISCC) of the canister, and canister lid and confinement welds and heat affected zones (HAZs) is managed by an AMP during the period of extended operation to ensure that no aging effect will result in the loss of their intended primary safety functions of confinement and structural integrity. Therefore, ready retrieval of the SNF is maintained during the period of extended operation by maintaining the structural integrity of the NAC-MPC canister to be lifted and transferred to a NAC-STC transport cask. During the AMR, the appropriate NAC-MPC canister components required for the ready retrieval of the SNF and/or canister have been identified as components required to maintain retrievability and identified as RE in the AMR tables in the CoC Renewal Application.

These efforts provide reasonable assurance that the SFAs will be capable of being removed from the canister by normal means or that the canister can be directly transferred to a certified NAC-STC transport cask for off-site transport.

3.7 OPERATING EXPERIENCE REVIEW AND PRE-APPLICATION INSPECTION RESULTS

3.7.1 Operating Experience

A review of available NAC-MPC System operating data has been performed to identify any off-normal, accident, or other event potentially effecting the performance of the NAC-MPC System. Based on the review performed of loading and storage operational data submitted by the three NAC-MPC System General Licensees, no normal operating events have been identified that would have a significant effect on overall system performance during the period of extended operation. There have been no off-normal, or accident events reported that would affect the safety functions of the in-scope SSCs.

3.7.2 Pre-Application Inspection Results

During the week of July 23 thru July 26, 2018, the Pre-Application Inspection of NAC-UMS System No. Vertical Concrete Cask (VCC) number 55 and Transportable Storage Canister (TSC) number 22 was performed at Maine Yankee (MY) in accordance with NAC Procedure Nos. 30013-P-01 and 30013-P-02. NAC International (NAC), MY and NAC's Nuclear Technology Users Group (NUTUG) collaborated on the performance of a pre-application inspection to support the NAC-UMS System and NAC-MPC System Certificate of Compliance (CoC) Renewal Applications.

The scope of the NAC-UMS System pre-application visual inspection program covered the following important to safety (ITS) systems, structures and components (SSCs):

- TSC accessible external surfaces;
- TSC accessible welds and heat affected zones (HAZs);
- Internal VCC accessible metallic components including inlets/outlets;
- External VCC accessible metallic components; and
- Reinforced VCC concrete structure

The purpose of the pre-application inspection was to demonstrate that the NAC-UMS System SSCs have not undergone unanticipated degradation during the initial 20-year certification period. The inspection results reported herein are intended to support the CoC Renewal Applications for both the NAC-UMS System and NAC-MPC System for an additional 40-year period of extended operations.

The MY VCC number 55 / TSC number 22 was selected for the pre-application inspection in accordance with the criteria of EPRI Technical Report, TR-3002005371 [3.9.15] as documented in NAC Technical Report No. ED20170046. TSC Rankings were assessed in accordance with EPRI CISCC Criteria, dated April 18, 2017 [3.9.365]. The assessment included an analysis of the bounding NAC-UMS System or NAC-MPC System from the combined system fleets of 302 deployed storage systems at seven (7) Independent Spent Fuel Storage Installation (ISFSI) sites located around the US. The MY NAC-UMS System selected was based on site location and conditions, cask heat load, and time in service. The NAC-UMS System selected for inspection at the MY ISFSI has a current heat load of < 4 kW and had been in service for almost 16 years at the time of inspection (placed into service on the MY ISFSI on 9/21/02).

3.7.2.1 Equipment and Personnel Qualification

A key component of the acceptability of the NAC-UMS System pre-application inspection results was to ensure that only qualified personnel are utilized and that the visual inspection procedure, including the capabilities of the remote visual inspection system, are demonstrated. The visual inspection procedure utilized for VT-1/VT-3 pre-application visual inspections was demonstrated in accordance with ASME Section XI, IWA-2000, 2007 Edition (Reference 14) and NAC Procedure No, 30013-P-02, Procedure for Visual Examination of NAC-UMS and NAC-MPC Dry Cask Storage System (Reference 2). The Visual Inspection Procedure was prepared and approved by an NDE Level III in accordance with ASME Code, Section XI, IWA-2000, 2007 Edition (Reference 14). The procedure specified the essential and non-essential variables.

The NDE Level III prepared Demonstration Report No. 30013-DM-001 (Reference 15) to document the procedure demonstration activities performed at Robotic Technology of Tennessee (RTT's) facility on May 15, 2018. The demonstration determined the capabilities of the GE Mentor Visual iQ Video Probe, provided by GE Inspection Technologies (GE-IT) for the remote VT-1 and VT-3 visual inspection and documented procedure parameters. The demonstration documents that the equipment meets the requirements of ASME Section XI, IWA-2211 for VT-1 and IWA-2213 for VT-3 (Reference 14). The demonstration report documents the qualification of the specifically identified visual inspection equipment to meet the specified Code requirements of Table IWA-2211-1 to resolve VT-1 and VT-3 characters from an EPRI Visual Examination Resolution Card.

Prior to the start of the remote visual inspections on-site, additional remote visual inspection equipment (enhanced technology) was introduced by GE-IT. The new equipment capabilities were documented in Demonstration Report No. 30013-DM-002 (Reference 21) and incorporated in NAC Procedure No. 30013-P-02 by PCN No. 30013-P-02-01, as required.

The GE-IT equipment demonstrated in support of NAC Procedure No. 30013-P-02 development was also used for the remote general visual inspections of the TSC shell and VCC interior components as described in NAC Procedure No. 30013-P-01.

Personnel performing the direct and remote VT-1 and VT-3 visual examinations of the TSC accessible welds/HAZs were certified in accordance with ASME Section XI, IWA-2300, 2007 Edition, as modified by 10CFR50.55, as required by NAC Procedure No. 30013-P-02.

Personnel performing direct and remote general visual inspections of TSC accessible external surfaces, internal VCC metallic surfaces, and external VCC metallic surfaces were qualified in accordance with ANSI/ASNT CP-189-1995 and had previous documented experience with visual examination of DCSS components.

Personnel performing direct and remote VT1/VT-3 and general visual inspections possessed current vision acuity test records in accordance with ASME Section XI, IWA-2320.

Personnel performing the reinforced VCC concrete structures monitoring were qualified based on experience, and had visual acuity acceptable to MY, the General Licensee, in accordance with the guidance of ACI 349.3 R-02.

3.7.2.2 NAC-UMS System Pre-Application Inspection Scope and Results

The following inspection sequences were performed during the NAC-UMS System pre-application inspection at MY on VCC #55/TSC #22. The specific sequence of inspections and identification of specific inspection findings are as documented in the completed Visual Examination Report, 30013-P-02 (Attachments 1 and 2) and the completed Pre-Application Traveler 30013-P-01 (Attachment 3).

Major items of inspection on the NAC-UMS System ITS SSCs and the identified inspection results are as follows:

Transportable Storage Canister (TSC):

The inspection scope for the TSC was as follows:

- Visual inspection (VT-3) using direct and/or remote means of 100% of accessible TSC longitudinal and circumferential welds and HAZs, the HAZ of the TSC shell to baseplate weld, external top five (5) inches of the top of the TSC corresponding to the HAZs of the shield and structural lid welds, the structural lid to shell weld and HAZ, and known areas of welded temporary supports or attachments subsequently removed and corresponding HAZs for atmospheric deposits, localized corrosion, and evidence of stress corrosion cracking (SCC). Target minimum coverage of the TSC confinement boundary welds and HAZs was established as 80% and actual results were approximately 85-90% of the TSC accessible weld and HAZ surfaces inspected.
- General visual inspection using direct and/or remote means of 100% of welded stainless-steel dry storage TSC confinement boundary readily accessible external surfaces including the top surfaces of the TSC structural lid, except for TSC weld areas and weld heat affected zones (HAZs) defined as two (2) inches either side of inspected welds, for atmospheric deposits, general condition of the TSC external surfaces, and localized corrosion. Potential areas of crevice corrosion such as areas of contact between the TSC and VCC liner, the bottom of the TSC and the top of the VCC pedestal, etc. were also inspected. Target minimum coverage of the TSC surfaces was established as 80% and actual results were approximately 95% of the TSC accessible surfaces inspected.

Parameters inspected and/or monitored for TSC surfaces included:

- Visual evidence of discontinuities and imperfections such as localized corrosion, including pitting corrosion, and stress corrosion cracking (SCC) of the TSC welds and weld HAZs.
- Size and location of localized corrosion and evidence of SCC on TSC welds and HAZs.
- Appearance and location of deposits on the TSC surfaces examined by general inspection
- The inspection criteria for the above inspection scope is identified in Section 7.1 of 30013-P-01

The sequence of TSC inspections and results were as follows:

- The first sequence of the pre-application inspection performed on July 23 was the remote video inspection of the TSC welds and HAZs. Inspection equipment was inserted into the VCC-TSC annulus through the outlet vents using the RTT robotic crawler. The crawler carrying the GE-IT video probe traveled along the VCC walls while viewing the TSC welds and HAZs. The results and acceptability of each of the welds and HAZs inspected are documented on Attachments 1 and 2 of Appendix E. The welds are identified as V-1 and V-2 for the two vertical shell welds, which were accessed from the East Vent. Welds identified as C-1 and C-2 are for the central circumferential weld and baseplate to shell weld, respectively.

- TSC shell longitudinal (axial) weld(s) and HAZs (V-1 and V-2) were inspected by remote VT-3 visual inspection methods. The inspection did not identify any indications of concern not meeting the acceptance criteria. The only issue identified was the presence of foreign material (e.g., bugs, moisture streaking, spider webbing, and small amounts of other debris).

- Weld V-1 showed an area of suspected interference with the Transfer Adapter Plate during TSC transfer into the VCC following loading and closure operations. The area shows residual white paint with some carbon staining around the scraped area from interaction with the carbon steel Transfer Adapter. The VT-3 inspection did not identify any depth to the scuffed area.

- TSC circumferential weld and HAZ, and baseplate to TSC shell circumferential weld and HAZ (C-1 and C-2, respectively) were inspected by VT-3 remote visual inspection on July 24. The inspection did not identify any indications of concern not meeting the acceptance criteria. The only issue identified was the presence of foreign material (e.g., bugs, moisture streaking, spider webbing, and small amounts of other debris). The remote visual inspection results for the C-1 and C-2 circumferential shell and baseplate welds and HAZs are documented in Appendix E.

- TSC structural lid weld and HAZ (T-1) and the top 5 inches of the TSC constituting the external circumferential HAZ (HAZ-1) corresponding to the internal shield lid and structural lid to the TSC shell welds were inspected by direct VT-3 visual examination on July 24. The direct visual inspection did not identify any indications of concern not meeting the acceptance criteria. The direct visual examination results for the upper 5 inches of the TSC shell defined as HAZ-1, and the structural lid weld and HAZ (T-1) are reported on Attachment 2 of Appendix E..

- During the TSC structural lid weld examination, Elcometer measurements were conducted on two locations on the TSC structural lid. Sample #1 reported results of 2.9 µg/cm2 and sample #3 reported results of 6.3 µg/cm2. These results are documented in the 30013-P-01 Inspection Traveler.

- TSC shell accessible surfaces outside of the TSC weld and HAZ areas were inspected by general visual inspection using direct and remote visual inspection methods including the top surfaces of the TSC structural lid on July 24 and the TSC shell surfaces on July 25. The inspection did not identify any significant indications exceeding the established acceptance criteria. The results of the inspections are reported on completed Inspection Traveler 30013-P-01, Appendix E.

<u>VCC Internal Metallic Inspection:</u>

The inspection scope for the VCC internal metallic components was as follows:

- A general visual inspection using direct observation and/or remote means for examining VCC coated internal metallic components of the VCC internal metallic components readily accessible surfaces for localized corrosion and pitting. The general condition of the VCC internal surfaces was examined for areas of unusual coating damage and atmospheric deposits. Target minimum coverage of the VCC interior metallic surfaces was established as 80% and actual results indicated approximately 95% of VCC accessible surfaces were inspected.
- The specific items inspected include the following VCC internal metallic components:
    - Underside of the VCC lid (using direct observation methods)
    - VCC top flange surfaces and seal gasket (using direct observation methods)
    - VCC shield plug and associated support ring surfaces (using direct observation methods)
    - VCC liner inner surfaces
    - Visible portions of the TSC pedestal plate
    - Exposed baffle weldment components
    - Inlet vent component surfaces
    - Outlet vent component surfaces

Parameters inspected and/or monitored for VCC internal coated steel surfaces included the following:

- Visual evidence of discontinuities, imperfections, and rust staining indicative of localized corrosion
- Pitting or crevice corrosion
- Unusual coating degradation
- Degradation of VCC lid seal gasket
- Blockage or obstruction of the VCC annulus and inlet and outlet vent openings

The results of the multiple direct and remote visual inspections were as follows:

- The VCC lid and VCC shield plug were removed on July 24 to allow access to the TSC Structural Lid for inspection of the structural lid weld and HAZ. The VCC lid and shield plug were in good condition with minimal paint deterioration. The VCC lid lifting threaded holes were chased with a tap to facilitate installation of the lifting hoist rings. The VCC lid gasket appeared in good condition. There were no identified conditions on the VCC lid, VCC shield plug, or support ring that exceeded the established acceptance criteria. Following inspection of the TSC structural lid weld and HAZ, and HAZ on outside top 5 inches of TSC, the VCC shield plug and lid were reinstalled following the placement of a new gaskets on the top flange surface.

- The remote general visual inspection of the interior VCC liner surfaces was performed on July 25 by the insertion of the RTT robotic crawler carrying the GE-IT video camera through the four upper outlets. Results of inspection identified limited deterioration of VCC liner coatings. The paint deterioration and localized corrosion (approximately 12 to 14 inches horizontally x 24 to 30 inches vertically detected during the south to east pass 1) of the liner surface were evaluated by MY Condition Report (CR) No. MY-CR-2018-124. As noted in the CR, NAC Calculation No. 30013-2002, Revision 0 concluded that localized corrosion of the VCC interior metallic surfaces was acceptable for the duration of the original certified period of 20 years plus the period of extended operation of 40 years. All outlet vents were unblocked and in good condition.

- The remote general visual inspection of the interior VCC lower metallic surfaces (e.g., pedestal, support plate and baseplate surfaces) was performed on July 25 by the insertion of the RTT robotic crawler carrying the GE-IT video camera through the four lower inlets. All the inlets vents were unblocked and all VCC lower interior structures were in excellent condition with no paint deterioration. The results of the VCC internal metallic components inspections are documented in completed Inspection Traveler 30013-P-01 Data Sheets.

External VCC Metallic Components Inspection:

The inspection scope for the VCC external metallic components was as follows:

- A general visual inspection of the VCC external metallic components using direct observation for general and localized corrosion, wear, cracking, areas of unusual coating damage, loss of preload (bolting), and general condition of the external VCC metallic component surfaces was performed on July 24 and July 25.

- The specific items inspected included the following VCC external metallic components:
  - Top surfaces of VCC lid
  - Top surfaces of VCC top flange
  - Exposed external surfaces of inlets and outlets including condition of inlet and outlet screens and attachments
  - Exposed VCC baseplate surfaces
  - VCC lid bolting and bolt holes

Parameters inspected and/or monitored for VCC external coated steel surfaces included the following:

- Visual evidence of general corrosion, discontinuities, imperfections, and/or significant rust staining indicative of corrosion, and wear
- Visual evidence of loose or missing bolts, physical displacement, and other conditions indicative of loss of preload
- Visual evidence of significant coating degradation (e.g., blisters, cracking, flaking, delamination) exceeding coating degradation levels previously identified and remediated during FSAR required annual maintenance inspections

The results of the inspections were as follows:

- The VCC external components were visually inspected, and no significant issues were identified.
- The threaded holes used for lifting the VCC lid contained residual corrosion products and were required to be chased prior to being able to insert swivel hoist lifting rings. The corrosion was a result of interaction between the VCC lid stainless steel bolts and the carbon steel VCC lid.
- Two new seals were installed on the VCC top flange, one on the ID and one on the OD of the bolt circle below the VCC lid prior to re-installation and bolt tightening.
- Inlet and outlet vent screens and attachments were found to be in good condition and required no corrective actions.

Results of the VCC external metallic components were documented in completed Inspection

Traveler 30013-P-01 Data Sheets

Reinforced VCC Concrete Structures Monitoring

The inspection scope for the reinforced VCC concrete structures monitoring were as follows:

- A general visual inspection by direct observation of the above-grade external VCC concrete surfaces that are directly exposed to outdoor air was performed on July 25
- The specific items inspected included the following concrete components and interfaces:
  - Readily accessible steel-to-concrete interfaces of the VCC bottom plate assembly (i.e., around the bottom end of the VCC and the openings of all four inlet vents) and all four outlet vent weldments
  - Readily accessible surfaces of all inlet and outlet screens and associated screen attachment hardware
  - VCC side and top concrete surfaces
  - Steel-to-concrete interfaces of the VCC top flange

Parameters inspected and/or monitored for significant VCC concrete structure aging effects exceeding the acceptance criteria included the following deterioration effects exceeding those previously identified and monitored during FSAR required annual maintenance inspections:

- Tier 3 cracking per ACI 349.3R-02
- Loss of material (spalling, scaling)
- Loss of bond to reinforcing steel observed by evidence of corrosion staining
- Significant porosity/permeability of concrete surfaces

The results of the inspections were as follows:

- No significant deterioration was identified during the VCC concrete monitoring that exceeded previously issues identified during required FSAR annual inspections and defined as Tier 2 cracks and minor Tier 2 spalling between the SE and NE inlets.

Results of the reinforced VCC concrete monitoring are documented in completed Inspection Traveler 30013-P-01 Data Sheets

3.2.4.3 NAC-UMS System Pre-Application Inspection Conclusions

Overall, the NAC-UMS System pre-application inspection of VCC #55 and TSC #22 resulted in no significant issues or inspection findings exceeding the established acceptance criteria except for the localized NITS VCC coating degradation on the VCC liner resulting in a minimal area of localized corrosion. The localized corrosion will have no long-term impact as VCC liner degradation has been evaluated in NAC Time Limited Aging Analysis (TLAA No. 30013-2002). This inspection finding was documented and evaluated in a Condition Report (CR) in the Inspection Report. The overall inspection of the VCC/TSC identified that the NAC-UMS System performed as expected during the nearly 16 years of deployment on the MY ISFSI, and no issues were identified that would result in future deterioration of the system’s ITS SSCs and subcomponents.

The preparation and review of supporting 10 CFR 72.48 determination regarding off-normal conditions related to the inspection requirements (e.g., temporary removal of the VCC lid and shield plug, potential use of temporary shielding, etc.) showed that the NAC-UMS System could be maintained in accordance with Limiting Condition of Operation (LCO) 3.1.6 including applicable conditions, required actions, and completion times during the limited duration of the inspection. In addition, the off-normal condition of the NAC-UMS System would not adversely affect the performance of the system to satisfy local and off-site dose limitations and protection of the stored TSC due to extreme weather or other environmental conditions. The 10 CFR 72.48 determination was supported by a shielding analysis evaluating local dose rates and site boundary effects of the VCC lid and shield plug removal would not exceed regulatory or site requirements. The thermal and structural performance of VCC #55 and TSC #22 were evaluated in NAC White Paper No. 30013-WP-01. The White Paper concluded that system thermal and structural performance would be maintained with appropriate safety margins. Site work control requests included actions to be taken in case of a severe weather event to preclude potential damage from tornado driven missile impacts. These reference documents provide a bases to prepare similar evaluations and 10 CFR 72.48 determinations for future inspections at all NAC-UMS System and NAC-MPC System ISFSIs.

Required FSAR inspections of the VCC and external metallic components will continue to be performed to monitor future performance of the VCC until the recertification of the NAC-UMS System and implementation of the proposed aging management program.

This first full inspection of the NAC-UMS System performed in accordance with the proposed AMPs showed the robustness of the design and effective operating performance of the system at an ISFSI located above the freeze line and selected as closest to a marine environment.

3.8 DESIGN BASIS DOCUMENT REVIEW

A complete documented review of all NAC-MPC System design bases documents has been performed to support the TLAA and AMP processes. A complete database of applicable NAC-MPC System design, licensing, and operating data was assembled to facilitate the review. Each individual document was reviewed to determine if it met the definition for a TLAA or impacted the safety function of the NAC-MPC System SSCs. The information gained from this review was utilized in the development of the TLAAs included with this renewal application, in the identification of operating environments and conditions, the identification of evaluated aging effects, and in the development of the identified Aging Management Programs.

None of the design basis documents reviewed affirmatively met the six questions identified in NUREG-1927 [3.9.2] as defining a TLAA. Each of the documents was reviewed against the six TLAA questions and the review and question response documented. A summary report of the Design Basis Document Review is provided in Appendix F.

The information gained from this review was utilized in the development of the TLAAs included with this renewal application, in the identification of operating environments and conditions, the identification of evaluated aging effects, and in the development of the identified Aging Management Programs.

3.9 REFERENCES

3.9.1 NAC International, Inc., "Final Safety Analysis Report for the NAC-MPC Multi-Purpose Canister System," Docket No. 72-1025,

- 3.9.1.a. NAC-MPC System FSAR, Revision 0, May 2000
- 3.9.1.b. NAC-MPC System FSAR, Revision 1, February 2002
- 3.9.1.c. NAC- MPC System FSAR, Revision 2, November 2002
- 3.9.1.d. NAC- MPC System FSAR, Revision 3, March 2004
- 3.9.1.e. NAC- MPC System FSAR, Revision 4, November 2004
- 3.9.1.f. NAC- MPC System FSAR, Revision 5, October 2005
- 3.9.1.g. NAC- MPC System FSAR, Revision 6, November 2006
- 3.9.1.h. NAC- MPC System FSAR, Revision 7, November 2008
- 3.9.1.i. NAC- MPC System FSAR, Revision 8, February 2009
- 3.9.1.j. NAC- MPC System FSAR, Revision 9, November 2010
- 3.9.1.k. NAC- MPC System FSAR, Revision 10, October 2012
- 3.9.1.l. NAC-MPC System FSAR, Revision 11, April 2018
- 3.9.1.m. NAC-MPC System FSAR, Revision 12, April 2019

3.9.2 NRC. NUREG-1927, Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance, Revision 1, June 2016

3.9.3 NEI. NEI 14-03, Revision 2, “Guidance for Operations-Based Aging Management for Dry Cask Storage,” December 2016.

3.9.4 NRC. NUREG-2214, “Managing Aging Processes in Storage (MAPS) Report”, Final Report, July 2019.

3.9.5 American Society for Testing and Materials (ASTM) C 1562-03

3.9.6 EPRI. Electric Power Research Institute (EPRI) Report TR-1003416, “Technical Bases for Extended Dry Storage of Spent Nuclear Fuel”, 2002

3.9.7 EPRI Report TR-108757, “Data Needs for Long-Term Dry Storage of Spent Fuel”, 1998

3.9.8 EPRI Report TR-1002882, “Spent Fuel Dry Storage Cask Inspection After Years of Operation”, 2011

3.9.9 International Atomic Energy Agency Technical Report Series No. 443, “Understanding and Managing Ageing of Material in Spent Fuel Storage Facilities”, 2006

3.9.10 NRC. NRC Interim Staff Guidance (ISG) 11, Revision 3, Cladding Considerations for the Transportation and Storage of Spent Fuel”, 2003

3.9.11 NRC. NUREG/CR-6745, “Dry Cask Storage Characterization Project – Phase 1: CASTOR V/21 Cask Opening and Examination”, 2001

3.9.12 NRC. NUREG/CR-6831, “Examination of PWR Fuel Rods after 15 Years in Dry Storage” 2003

3.9.13 NRC. NUREG-1522, “Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures”, 1995

3.9.14 NRC. NUREG-1801, R2, “Generic Aging Lessons Learned (GALL) Report”, 2010

3.9.15 EPRI Technical Report, TR-3002005371, “Susceptibility Assessment Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless-Steel Canisters for Dry Storage Systems”, 2015

3.9.16 EPRI Technical Report, TR-3002008193, “Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless-Steel Canisters”, 2017.

3.9.17 EPRI Technical Report Update. TR-3002002785, “Failure Modes and Effects Analysis (FEMA) of Welded Stainless Steel Dry Cask Storage Systems” 2013

3.9.18 NRC. NRC Interim Staff Guidance (ISG)-2, Revision 2, “Fuel Retrievability in Spent Fuel Storage Applications”, 2016

3.9.19 NRC. NUREG/CR-7170, “Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Chloride and Non-Chloride Salts”, 2014

3.9.20 NRC. NRC Report, “Identification and Prioritization of the Technical Information Needs Affecting Potential Regulation of Extended Storage and Transport of Spent Nuclear Fuel”, 2014

3.9.21 DOE/ANL Report ANL-12/29, “Managing Aging Effects on Dry Cask Storage Systems for Extended Long-Term Storage and Transportation”, 2012

3.9.22 NRC. NRC Information Notice 2011-20, “Concrete Degradation by Alkali-Silica Reaction NRC Interim Staff Guidance”, 2011

3.9.23 NRC. NRC Interim Staff Guidance (ISG)-24, Revision 0, “The Use of a Demonstration Program as a Surveillance Tool for Confirmation of Integrity for Continued Storage of High Burnup Fuel Beyond 20 Years”, 2014

3.9.24 NRC. NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36”, 1980

3.9.25 ANSI N14.6, “Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500kg) or More for Nuclear Materials”

3.9.26 ANSI N14.5, “Radioactive Materials – Leakage Tests on Packages for Shipment”, 2014

3.9.27 AISC. ANSI/AISC 360-10, “Specification for Structural Steel Buildings.” Chicago, Illinois: American Institute of Steel Construction. 2010.

3.9.28 Alexander, D.J. “Effects of Irradiation on the Mechanical Properties of 6061-T651 Aluminum Base Metal and Weldments.” ASTM Special Technical Publication. Vol. 1325. 42 pp. 1,027–1,044. 1999.

3.9.29 Alexander, D.J. and R.K. Nanstand. “The Effects of Aging for 50,000 Hours at 343°C on the Mechanical Properties of Type 308 Stainless Steel Weldments.” Proceedings of the Seventh International Symposium on Environmental Degradation of Materials in Nuclear Power Systems–Water Reactors. Breckenridge, Colorado. NACE. Houston, Texas. pp. 747–758. 1995.

3.9.30 Alexander, D., P. Doubell, and C. Wicker. “Degradation of Safety Injection Systems and Containment Spray Piping and Tank—Fracture Toughness Analysis.” Presentation at Fontevraud 7, *Contribution of Materials Investigations to Improve the Safety and Performance of LWRs*, September 26–30, 2010. Avignon, France. 2010.

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

3.9.32 Andresen, P.L., F.P. Ford, K. Gott, R.L. Jones, P.M. Scott, T. Shoji, Staehle, and R.L. Tapping. “Expert Panel Report on Proactive Materials Degradation Assessment.” NUREG/CR-6923. Washington, DC: U.S. Nuclear Regulatory Commission. 2007.

3.9.33 ASM International. *Metals Handbook, Desk Edition, Second Edition*. Materials Park, Ohio: ASM International. pp. 280–285. 1998.

3.9.34 ASM International. “Heat Treating of Aluminum Alloys.” In ASM Handbook, Vol. 4, Heat *Treating*. Materials Park, Ohio: ASM International. pp. 841–879. 1991.

3.9.35 ASME. Boiler and Pressure Vessel (B&PV) Code, Section III, “Rules for Construction of Nuclear Facility Components,” Division 1, Subsection NB, “Class 1 Components,” and Subsection NC, “Class 2 Components”; American Society of Mechanical Engineers. 2007a.

3.9.36 ASME. Boiler and Pressure Vessel (B&PV) Code, Section II, “Materials,” Part D, “Properties,” American Society of Mechanical Engineers. 2007b.

3.9.37 Baboian, R. “Galvanic Corrosion.” In ASM Handbook, Vol. 13A, Corrosion*: Fundamentals, Testing, and Protection.* Materials Park, Ohio: ASM International. pp. 210–213. 2003.

3.9.38 Baggerly, R. “Environmental Failures of High Strength Bolts, in Case Histories on Integrity and Failures in Industry.” V., ed. *Proceedings of an International Symposium on Case Histories on Integrity and Failures in Industry*, September 28–October 2, 1999. Milan, Italy. 1999.

3.9.39 Bass, H.K. “The Corrosion of Aluminum in Boric Acid Solutions.” Master’s thesis. Agricultural and Mechanical College of Texas. College Station, Texas. 1956.

3.9.40 Basson, J.P. and C. Wicker. “Environmentally Induced Transgranular Stress Corrosion Cracking of 304L Stainless Steel Components at Koeberg.” Fontevraud 5 International Symposium*, Contributions of Materials Investigations to Resolution of Problems Encountered in Pressurized Water Reactors*. Société Française d’Energie Nucléaire–SFEN. Paris, France. Vol. 1–2. 1,175p. September 2002.

3.9.41 Baumgattner, M. and H. Kaesche. “The Nature of Crevice Corrosion of Aluminum in Chloride Solutions.” *Werkstoffe und Korrosion.* Vol. 39. pp. 129–135. 1988.

3.9.42 Bickford, J.H. *Introduction to the Design and Behavior of Bolted Joints.* 4th Edition. Boca Raton, Florida: CRC Press. 2008.

3.9.43 Blau, P.J. “Rolling Contact Wear.” In ASM Handbook Vol. 18, *Friction, Lubrication, and Wear Technology*. Materials Park, Ohio: ASM International. pp. 257–262. 1992.

3.9.44 Blewitt, T.H., R.R. Coltman, C.E. Klabunde, and T.S. Noggle. “Low-Temperature Reactor Irradiation Effects in Metals.” *Journal of Applied Physics*. Vol. 28. pp. 639–644. 1957.

3.9.45 Bruhn, D.F., S.M. Frank, F.F. Roberto, P.J. Pinhero, and S.G. Johnson. “Microbial Biofilm Growth on Irradiated, Spent Nuclear Fuel Cladding. *Journal of Nuclear Materials*. Vol. 384, No. 2. pp. 140–145. 2009.

3.9.46 Cadek, J. *Creep of Metallic Materials*. Elsevier Science Publishing Company, Inc. 1988.

3.9.47 Caprio, J.J., A. Parra, and L. Martinez. “Scanning Electron Microscopy and Infrared Spectroscopic Studies of Marine Atmospheric Corrosion Products of Steel.” Paper No. 242. Houston, Texas: NACE International. 1995.

3.9.48 Caseres, L. “Electrochemical Behavior of Aluminized Steel Type 2 in Scale-forming Waters.” Ph.D. dissertation. Tampa, Florida: University of South Florida. 2007.

3.9.49 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.

3.9.50 Chandra, K., K. Vivekanand, V.S. Raja, R. Tewari, and G.K. Dey. “Low Temperature Thermal Ageing Embrittlement of Austenitic Stainless-Steel Welds and its Electrochemical Assessment.” *Corrosion Science*. Vol. 54. pp. 278–290. 2012.

3.9.51 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.

3.9.52 Code of Federal Regulations. Title 10, Energy,” Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix H, “Reactor Vessel Material Surveillance Program Requirements.” Washington, DC: U.S. Government Printing Office. 2015.

3.9.53 Cohen, A. “Corrosion of Copper and Copper Alloys.” In ASM Handbook, Vol. 13B, *Corrosion: Materials*. Materials Park, Ohio: ASM International. pp. 125–163. 2005.

3.9.54 Cook, A., J. Duff, N. Stevens, S. Lyon, A. Sherry, and T.J. Marrow. “Preliminary Evaluation of Digital Image Correlation for *In-Situ* Observation of Low Temperature Atmospheric-Induced Chloride Stress Corrosion Cracking in Austenitic Stainless Steels.” *ECS Transactions*. Vol. 25, No. 37. pp. 119–132. 2010.

3.9.55 Crook, P. “Corrosion of Nickel and Nickel-Base Alloys.” In ASM Handbook, Vol. 13B, *Corrosion: Materials*. Materials Park, Ohio: ASM International. pp. 228–251. 2005.

3.9.56 David, D., C. Lemaitre, and C. Crusset. “Archaeological Analogue Studies for the Prediction of Long-Term Corrosion on Buried Metals.” D. Feron and D. D. Macdonald, eds. EFC Series Vol. 36, *Prediction of Long-Term Corrosion Behavior in Nuclear Waste Systems*. 242p. Maney, London, United Kingdom. European Federation of Corrosion Publications. 2002.

3.9.57 Davison, R.M., T. DeBold, and M.J. Johnson. “Corrosion of Stainless Steels.” In ASM Handbook Vol. 13, *Corrosion*. Materials Park, Ohio: ASM International. pp. 547–565. 1987.

3.9.58 Dexter, S.C. “Microbiologically Influenced Corrosion.” In ASM Handbook, Vol. 13A, Corrosion*: Fundamentals, Testing, and Protection*. Materials Park, Ohio: ASM International. pp. 398–416. 2003.

3.9.59 Dragun, J. “The Soil Chemistry of Hazardous Materials.” Silver Spring, Maryland: Hazardous Materials Control Research Institute. pp. 325–445. 1988.

3.9.60 Earthman, J.C. “Introduction to Creep and Stress-Relaxation Testing.” In ASM Handbook. Vol. 8, *Mechanical Testing and Evaluation*. Materials Park, Ohio: ASM International. pp. 361–362. 2000.

3.9.61 EPRI. “Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications,” Report 1019110. Palo Alto, California: Electric Power Research Institute. 2009.

3.9.62 EPRI. “Guideline on Nuclear Safety-Related Coatings,” Revision 2, Report 1019157.Palo Alto, California: Electric Power Research Institute. 2009.

3.9.63 EPRI. “Aging Effects for Structures and Structural Components (Structural Tools).” Report 1015078. Palo Alto, California: Electric Power Research Institute. 2007.

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

3.9.65 EPRI. “Effects of Marine Environments on Stress Corrosion Cracking of Austenitic Stainless Steels.” Report 1011820. Palo Alto, California: Electric Power Research Institute. 2005.

3.9.66 Fabritsiev, S.A., A.S. Pokrovsky, and S.E. Ostrovsky. "Effect of the Irradiation–Annealing– Irradiation Cycle on the Mechanical Properties of Pure Copper and Copper Alloy." *Journal of Nuclear Materials*. Vol. 324. pp. 23–32. 2004.

3.9.67 Ferrell, K., "Assessment of Aluminum Structural Materials for Service Within the ANS Reflector Vessel," ORNL/TM-13049, Oak Ridge National Laboratory, August 1995.

3.9.68 Farrell, K. and R.T. King. "Radiation-Induced Strengthening and Embrittlement in Aluminum." *Metallurgical Transactions A. Physical Metallurgy and Materials Science*. Vol. 4, Issue 5. pp. 1,223–1,231. 1973.

3.9.69 Farro, N.W., L. Veleva, and P. Aguilar. "Copper Marine Corrosion: I. Corrosion Rates in Atmospheric and Seawater Environments of Peruvian Port." *The Open Corrosion Journal.* Vol. 2. pp. 130–138. 2009.

3.9.70 Feliu, S., M. Morcillo, and S. Feliu, Jr. "The Prediction of Atmospheric Corrosion from Meteorological and Pollution Parameters-II, Long-Term Forecasts." *Corrosion Science*. Vol. 34, No. 3. pp. 415–422. 1993.

3.9.71 Foct, F. and J.-M. Gras. "Semi-Empirical Model for Carbon Steel Corrosion in Long Term Geological Nuclear Waste Disposal." D. Feron and D.D. Macdonald, eds. EFC Series. Vol. 36. *Prediction of Long-Term Corrosion Behavior in Nuclear Waste Systems*. Maney, London, United Kingdom. 91p. 2002.

3.9.72 Foley, R.T. "Localized Corrosion of Aluminum Alloys—A Review." *Corrosion*. Vol. 42. 9, pp. 277–288. 1986.

3.9.73 Fonseca, I.T.E., R. Picciochi, M.H. Mendonca, and A.C. Ramos. "The Atmospheric Corrosion of Copper at Two Sites in Portugal: A Comparative Study." *Corrosion Science*. Vol. 46. 12 pp. 547–561. 2004.

3.9.74 FPL. "Turkey Point Nuclear Plant Unit 3, Docket No. 50-250, 10 CFR 50.55a, Request for Temporary Non-Code Repair, Spent Fuel Pool Cooling Line." Florida Power and Light. ADAMS Accession No ML052780060. 2005.

3.9.75 Frankel. G.S. "Pitting Corrosion." In ASM Handbook, Vol. 13A, Corrosion*: Fundamentals, Testing, and Protection*. Materials Park, Ohio: ASM International. pp. 236–241. 2003.

3.9.76 NRC, "Study of the ACI 349.3R-02 Tier 2 (i.e., Section 5.2.1) Criteria Impacts on Dose Rates for Several Spent Nuclear Fuel Dry Storage System Designs."

3.9.77 Gamble, R. "BWRVIP-100-A: BWR Vessel and Internal Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds." EPRI-1013396. Palo Alto, California: Electric Power Research Institute. 2006.

3.9.78 Garcia-Guinea, J., V. Cardenes, A.T. Martınez, and M.J. Martınez. "Fungal Bioturbation Paths in a Compact Disk." *Naturwissenschaften* (*The Science of Nature*). Vol. 88. pp. 351–354. 2001.

3.9.79 Gavendra, D.J., W.F. Michaud, T.M. Galvin, W.F. Burke, and O.K. Chopra. NUREG/CR–6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds." Washington, DC: U.S. Nuclear Regulatory Commission. May 1996.

3.9.80 Ghali, E. "Aluminum and Aluminum Alloys." In *Uhlig's Corrosion Handbook.* 3rd Edition. R.W. Revie, eds. John Wiley & Sons, Inc. pp. 715–745. 2011.

3.9.81 Ghali, E. *Corrosion Resistance of Aluminum and Magnesium Alloys Understanding*, *Performance, and Testing*. Hoboken, New Jersey: John Wiley & Sons, Inc. 2010.

3.9.82 Gibeling, J.C. "Creep Deformation of Metals, Polymers, Ceramics, and Composites." In ASM Handbook, Vol. 8, *Mechanical Testing and Evaluation*. Materials Park, Ohio: ASM International. pp. 363–368. 2000.

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

3.9.84 Hack, H.P. *Galvanic Corrosion Test Methods*. Houston, Texas: NACE International. 1993.

3.9.85 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." FCRD-USED-2011-000136. Rev. 0, PNNL-20509Richland, Washington: Pacific Northwest National Laboratory. 2012.

3.9.86 He, X. D. Dunn. "Crevice Corrosion Penetration Rates of Alloy 22 in Chloride-Containing Waters." *Corrosion*. Vol. 63. pp. 145–158. 2007.

3.9.87 Herman, R.S. and A.P. Castillo. ASTM-STP 558, "Short-Term Atmospheric Corrosion of Various Copper-Base Alloys—Two- and Four-Year Results." West Conshohocken, Pennsylvania: ASTM International. pp. 82–96. 1974.

3.9.88 Hoeppner, D.W. "Industrial Significance of Fatigue Problems." In ASM Handbook, Vol. 19. *Fatigue and Fracture*." Materials Park, Ohio: ASM International. pp. 3–4. 1996.

3.9.89 Horn, J.M. and A. Meike. "Microbial Activity at Yucca Mountain." UCRL-ID-122256. Livermore, California: Lawrence Livermore National Laboratory. 1995.

3.9.90 Hosler, R. "Screening Criteria for ID and OD-Initiated SCC of Pressure Boundary Stainless Steel Components (Phase 1 of I&E Guideline Development)." AREVA document 51-9142337-000. October 18, 2010.

3.9.91 Jack, T.R., M.J. Wilmott, R.L. Sutherby, and R.G. Worthingham. "External Corrosion of Line Pipe—A Summary of Research Activities." *Materials Performance*. Vol. 35. pp. 18–24. 1996.

3.9.92 Jones, R.H. “Stress corrosion Cracking.” In ASM Handbook, Vol. 13A, *Corrosion: Fundamentals, Testing, and Protection*. Materials Park, Ohio: ASM International. pp. 346–366. 2003.

3.9.93 Jones, R.H. *Stress corrosion Cracking*. Materials Park, Ohio: ASM International: 1992.

3.9.94 Jung, H., P. Shukla, T. Ahn, L. Tipton, K. Das, X. He, and D. Basu. “Extended Storage and Transportation: Evaluation of Drying Adequacy.” San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2013.

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

3.9.96 Kaufman, J.G. *Properties of Aluminum Alloys: Tensile, Creep, and Fatigue Data at High and Low Temperatures*. Materials Park, Ohio. ASM International. 1999.

3.9.97 Kelly, R.G. “Crevice Corrosion.” In ASM Handbook, Vol. 13A, *Corrosion: Fundamentals, Testing, and Protection*. Materials Park, Ohio: ASM International. pp. 242–247. 2003.

3.9.98 Kim, S. and Y. Kim. “Estimation of Thermal Aging Embrittlement of LWR Primary Pressure Boundary Components.” *Journal of the Korean Nuclear Society*. Vol. 30, No. 6. pp. 609–616. 1998.

3.9.99 King F. “Microbiologically Influenced Corrosion of Nuclear Waste Containers.” *Corrosion*. Vol. 65. pp. 233–251. 2009.

3.9.100 Kodama, T. “Corrosion of Wrought Carbon Steels.” In ASM Handbook, Vol. 13B, *Corrosion: Materials*. Materials Park, Ohio: ASM International. pp. 5–10. 2005.

3.9.101 Krauss, G., *Steels: Processing, Structure, and Performance.* Materials Park, Ohio. ASM International. pp. 396–402. 2005.

3.9.102 Kulak, G.L, J.W. Fisher, and J.H.A. Struik. *Guide to Design Criteria for Bolted and Riveted Joints*. 2nd ed. Chicago, Illinois: AISC Inc. 2001.

3.9.103 Leidheiser, H. *The Corrosion of Copper, Tin, and Their Alloys*. New York, New York: John Wiley & Sons, Inc. 1974.

3.9.104 Lillard, J.A. and R.J. Hanrahan, Jr. “Corrosion of Uranium and Uranium Alloys, Corrosion: Materials.” Vol 13B, ASM Handbook. ASM International. pp. 370–384. 2005.

3.9.105 Little, B.J. and P.A. Wagner. “An Overview of Microbiologically Influenced Corrosion of Metals and Alloys Used in the Storage of Nuclear Wastes.” *Canadian Journal of Microbiology*. Vol. 42. pp. 367–374. 1996.

3.9.106 Little, B.J. and J.S. Lee. “Microbiologically Influenced Corrosion.” U.S. Naval Research Laboratory Report NRL/BC/7303-08-8209. 2009.

3.9.107 Magee, J.H. "Wear of Stainless Steels." In ASM Handbook, Vol. 18, *Friction, Lubrication, and Wear Technology*. Materials Park, Ohio: ASM International. pp. 710–724. 1992.

3.9.108 Manaktala, H.K. "Degradation Modes in Candidate Copper-Based Materials for High-Level Radwaste Canisters." *Corrosion/90*. Paper No. 512. Las Vegas, Nevada: NACE. 1990.

3.9.109 Maruthamuthu, S., N. Muthukumar, M. Natesan, and N. Palaniswamy. "Role of Air Microbes on Atmospheric Corrosion." *Current Science*. Vol. 94. pp. 359–363. 2008.

3.9.110 Mayuzumi, M., J. Tani, and T. Arai. "Chloride Induced Stress Corrosion Cracking of Candidate Canister Materials for Dry Storage of Spent Fuel." *Nuclear Engineering and Design*. Vol. 238, No. 5. pp. 1,227–1,232. 2008.

3.9.111 McCuen, R.H. and P. Albrecht. "Composite Modeling of Atmospheric Corrosion Penetration Data." STP 1194, *Application of Accelerated Corrosion Testing to Service Life Prediction of Materials.* ASTM International. West Conshohocken, Pennsylvania. 1994.

3.9.112 McMahon, C.J. "Hydrogen-Induced Intergranular Fracture of Steels." *Engineering Fracture Mechanics*. Vol. 68. pp. 773–788. 2001.

3.9.113 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.

3.9.114 Morgan, J.D. "Report on Relative Corrosivity of Atmospheres at Various Distances from the Seacoast." NASA Report MTB 099-74. National Aeronautics and Space Administration. Cape Canaveral, Florida: Kennedy Space Center. 1980.

3.9.115 Morrison, J.D. "Corrosion Study of Bare and Coated Stainless Steel." NASA TN D-6519. Washington, DC: National Aeronautics and Space Administration. 1972.

3.9.116 Munier, G.B., L.A. Psota-Kelty, and J.D. Sinclair. *Atmospheric Corrosion*. W.H. Ailor, ed. Wiley-Interscience. New York, New York. 275p. 1982.

3.9.117 NACE. *Corrosion Engineer's Reference Book*. Third Edition. Edited by R. Baboian. Houston, Texas: NACE International. 2002.

3.9.118 Nguyen, T.H. and R.T. Foley. "On the Mechanism of Pitting of Aluminum." *Journal of Electrochemical Society*. Vol. 126. pp. 1,855–1,860. 1979.

3.9.119 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.

3.9.120 NRC. "Finite Element Analysis of Weld Residual Stresses in Austenitic Stainless Steel Dry Cask Storage System Canisters." NRC Technical Letter Report. Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML13330A512. 2013.

3.9.121 NRC. “Potential Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canisters.” NRC Information Notice 2012-20. Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML12319A440. 2012.

3.9.122 NRC. NUREG–1536, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility.” Rev. 1. Washington, DC: U.S. Nuclear Regulatory Commission. 2010.

3.9.123 NRC. “Outside Diameter Initiated Stress Corrosion Cracking Revised Final White Paper.” PA–MSC–0474.” Letter (October 14) to NRC From M.L. Arey, Jr. (PWROG Owners Group). Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML110400241. 2010.

3.9.124 NRC. Regulatory Guide 1.54, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” Rev. 2. Washington, DC: U.S. Nuclear Regulatory Commission. 2010.

3.9.125 NRC. “Failure of Control Rod Drive Mechanism Lead Screw Male Coupling at a Babcock and Wilcox-designed Facility.” NRC Information Notice 2007-02. Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML070100459. 2007.

3.9.126 NRC. NUREG–1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” Rev. 0. Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML003686776. 2000.

3.9.127 NRC. “ECCS Suction Header Leaks Result in Both ECCS Trains Inoperable and TS 3.0.3 Entry.” Licensee Event Report 1999-003-00. ADAMS Legacy Library Accession No. 9905130085. Washington, DC: U.S. Nuclear Regulatory Commission. April 1999.

3.9.128 Nuclear Decommissioning Authority. “Literature Review of Atmospheric Stress Corrosion Cracking of Stainless Steels Report to Nirex.” Report No. NR3090/043. Cumbria, United Kingdom: Nuclear Decommissioning Authority. 2007.

3.9.129 NWTRB. “Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel.” Washington, DC: Nuclear Waste Technical Review Board. 2010.

3.9.130 Odette, G.R. and G.E. Lucas. “Embrittlement of Nuclear Reactor Pressure Vessels.” *Journal of Metals*. Vol. 53, Issue 7. pp.18-22. 2001.

3.9.131 Olender, A., J. Gorman, C. Marks, and G. Ilevbare. “Recent Operating Experience Issues with 17-4 PH in LWRs.” Fontevraud 8: Conference on Contribution of Materials Investigations and Operating Experience to LWRs’ Safety, Performance and Reliability. France. 2015.

3.9.132 Parra, A., J. Carpio, and L. Martinez. “Microbial Corrosion of Metals Exposed to Air in Tropical Marine Environments.” *Materials Performance*. Vol. 35. pp. 44–50. 1996.

3.9.133 Phull, B. "Evaluating Stress Corrosion Cracking." In ASM Handbook, Vol. 13A, *Corrosion: Fundamentals, Testing, and Protection*. Materials Park, Ohio: ASM International. pp. 575–616. 2003.

3.9.134 Phull, B. "Evaluating Uniform Corrosion." In ASM Handbook, Vol. 13A, *Corrosion: Fundamentals, Testing, and Protection*. Materials Park, Ohio: ASM International. pp. 542–544. 2003.

3.9.135 Pourbaix, M. *Atlas of Electrochemical Equilibria in Aqueous Solutions*. 2nd ed. Houston, Texas: NACE. 1974.

3.9.136 Revie, R.W. *Uhlig's Corrosion Handbook.* Second Edition. Hoboken, New Jersey: John Wiley and Sons. 2000.

3.9.137 Rowcliffe, A.F., L.K. Mansur, D.T. Hoelzer, and R.K. Nanstad. "Perspectives on Radiation Effects in Nickel-Base Alloys for Applications in Advanced Reactors." *Journal of Nuclear Materials*. Vol. 392. pp. 341–352. 2009.

3.9.138 Rozenfeld, I.L. "Atmospheric Corrosion of Metals." Houston, Texas: NACE. 1972.

3.9.139 Sachs, K. and D.G. Evans. "The Relaxation of Bolts at High Temperatures." Report C364/73. Wolverhampton, United Kingdom: GKN Group Technological Center. 1973.

3.9.140 Samuels, I.E. *Metals Engineering: A Technical Guide*. Metals Park, Ohio: ASM International. 1988.

3.9.141 Shirai, K., J. Tani, T. Arai, M. Wataru, H. Takeda, and T. Saegusa. "SCC Evaluation Test of a Multi-Purpose Canister." Presentation at the *13th International High-Level Radioactive Waste Management Conference*, Albuquerque, New Mexico, April 10–14, 2011. LaGrange Park, Illinois: American Nuclear Society. 2011.

3.9.142 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.

3.9.143 Summerson, T.J., M.J. Pryor, D.S. Keir, and R.J. Hogan. "Pit Depth Measurements as a Means of Evaluating the Corrosion Resistance of Aluminum in Seawater." ASTM STP 196.pp. 157–175. West Conshohocken, Pennsylvania: ASTM International. 1957.

3.9.144 Tani, J.I., M. Mayuzurmi, and N, Hara. "Initiation and Propagation of Stress Corrosion Cracking of Stainless Steel Canister for Concrete Cask Storage of Spent Nuclear Fuel." *Corrosion*. Vol. 65, No. 3. pp. 187–194. 2009.

3.9.145 Tator, K.B. "Degradation of Protective Coatings." Corrosion: Materials. Vol 13B. ASM Handbook. ASM International. pp. 589–599. 2005.

3.9.146 Tracy, A.W. "Effect of Natural Atmospheres on Copper Alloys: 20 Year Test." *Atmospheric Corrosion of Nonferrous Metals*. ASTM-STP 175. 67p. West Conshohocken, Pennsylvania: ASTM International. 1955.

3.9.147 Vargel, C. *Corrosion of Aluminum.* San Diego, California: Elsevier, Inc. 2004.

3.9.148 van Bodegom, L., K. van Gelder, M.K.F. Paksa, and L. van Raam. “Effect of Glycol and Methanol on CO_2 Corrosion of Carbon Steel.” *Proceeding of CORROSION Conference*. Paper No. 55. Houston, Texas: NACE International. 1987.

3.9.149 Walch, M. and R. Mitchell. “The Role of Microorganisms in Hydrogen Embrittlement of Metals.” *Proceeding of CORROSION Conference*. Paper No. 249. Houston, Texas: NACE International. 1983.

3.9.150 Was, G.S., J. Busby, and P.L. Andresen. “Effect of Irradiation on Stress Corrosion Cracking and Corrosion in Light Water Reactors.” In ASM Handbook, Vol. 13C, *Corrosion: Environments and Industries*. Materials Park, Ohio: ASM International. pp. 386–414. 2006.

3.9.151 West, G.A. and C.D. Watson. “Gamma Radiation Damage and Decontamination Evaluation of Protective Coatings and Other Materials for Hot Laboratory and Fuel Processing Facilities.” ORNL-3589. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1965.

3.9.152 NRC Certificate of Compliance for NAC-STC Transport Cask, Docket 71-9235, CoC No. 9253, Revision 22, July 8, 2019.

3.9.153 ASTM International. “Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application.” ASTM A887–89. West Conshohocken, Pennsylvania: ASTM International. 2009.

3.9.154 NRC. NUREG/CR-7171, “DA Review of the Effects of Radiation on Microstructure and Properties of Concretes Used in Nuclear Power Plants”, Revision 0, November 2013

3.9.155 ACI 349.3R-02, “Evaluation of Existing Nuclear Safety-Related Concrete Structures”

3.9.156 EPRI. “Strategy for Managing the Long-Term Use of BORAL® in Spent Fuel Storage Pools.” Report 1025204. Palo Alto, California: Electric Power Research Institute. 2012.

3.9.157 EPRI. “BORAL® Behavior Under Simulated Cask Vacuum Drying. Part 2 Test Results.” Report 1009696. Palo Alto, California: Electric Power Research Institute. 2004.

3.9.158 Holtec International. “Final Safety Analysis Report for the HI-STORM 100 Cask System, Revision 12.” Holtec Report No. HI-2002444. USNRC Docket No. 72-1014. pp. 1.2-18. ADAMS Accession No. ML14086A410. 2014.

3.9.159 Robino, C.V. and M.J. Cieslak. “Fusion Welding of a Modern Borated Stainless Steel.” *Welding Journal.* Vol. 76, No. 1. pp. 11-s – 23-s. 1997.

3.9.160 Soliman, S.E., D.L. Youchison, A.J. Baratta, and T.A. Ballrelt. “Neutron Effects on Borated Stainless Steel.” *Nuclear Technology*. Vol. 96. pp. 346–352. 1991.

3.9.161 BISCO Products, Inc. “NS-3 Specification Sheet.” (ADAMS ML110730731), June 23, 1986.

3.9.162 Cota, S.S., V. Vasconcelos, M. Senne, Jr., L.OL. Carvalho, D.B. Rezende, and R.F. Cõrrea. “Changes in Mechanical Properties Due to Gamma Irradiation of High-Density Polyethylene.” Brazilian Journal of Chemical Engineering. Volume 24, No. 02. pp. 259–265. 12 2007.

3.9.163 Fu, L., R.A. Fouracre, and H.M. Banford. “An Investigation of Radiation Damage in Cured Epoxy Resin System Using Regression Experiment Design, Electrical Insulation and Dielectric Phenomena.” 1988 Annual Report, Conference on Electrical Insulation and Dielectric Phenomena. IEEE Dielectrics and Electrical Insulation Society. 1988.

3.9.164 McManus, H.L. and C.C. Chamis. “Stress and Damage in Polymer Matrix Composite Materials Due to Material Degradation at High Temperatures.” NASA Technical Memorandum 4682. Cambridge, Massachusetts: Massachusetts Institute of Technology. 1996.

3.9.165 NRC. “Safety Evaluation Report for License Renewal: Calvert Cliffs Nuclear Power Plant Independent Spent Fuel Storage Installation.” Washington, DC: U.S. Nuclear Regulatory Commission. 2014b.

3.9.166 NRC. “Safety Evaluation Report for License Renewal: Surry Independent Spent Fuel Storage Installation.” Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. 5 ML050590266. 2005.

3.9.167 ACI. ACI 305R-10, “Guide to Hot Weather Concreting.” Farmington Hills, Michigan: American Concrete Institute. 2010.

3.9.168 ACI. ACI 221.1R-98, “State-of-the-Art Report on Alkali-Aggregate Reactivity.” Farmington Hills, Michigan: American Concrete Institute. 2008a.

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

3.9.170 ACI. ACI 201.2R-08, “Guide to Durable Concrete.” Farmington Hills, Michigan: American Concrete Institute. 2008.

3.9.171 ACI. ACI 308R-01, “Guide to Curing Concrete.” Farmington Hills, Michigan: American Concrete Institute. 2008.

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

3.9.173 ACI. ACI 318-05, “Building Code Requirements for Structural Concrete and Commentary.” Farmington Hills, Michigan: American Concrete Institute. 2005.

3.9.174 ACI. ACI 221.1R-98, “State-of-the-Art Report on Alkali-Aggregate Reactivity.” Farmington Hills, Michigan: American Concrete Institute. 1998.

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

3.9.176 Akiyoshi, M. “Thermal Diffusivity of Ceramics at the Neutron Irradiation Temperature Estimated from Post-Irradiation Measurements at 123–413 K.” *Journal of Nuclear Materials*. Vol. 386–388. pp. 303–306. 2009.

3.9.177 Akiyoshi, M. and T. Yano. “Neutron-Irradiation Effect in Ceramics Evaluated from Macroscopic Property Changes in As-Irradiated and Annealed Specimens.” *Progress in Nuclear Energy*. Vol. 50. pp. 567–574. 2008.

3.9.178 Akiyoshi, M., I. Takagi, T. Yano, N. Akasaka, and Y. Tachi. “Thermal Conductivity of Ceramics During Irradiation.” *Fusion Engineering and Design*. Vol. 81. pp. 321–325. 2006.

3.9.179 ASME. “ASME Boiler and Pressure Vessel Code, Section III, Division 2.” New York, New York: American Society of Mechanical Engineers. 2007.

3.9.180 ASME. “ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL.” New York, New York: American Society of Mechanical Engineers. 1995.

3.9.181 ASTM International. ASTM C33, “Standard Specification for Concrete Aggregates.” West Conshohocken, Pennsylvania: American Society for Testing and Materials. 2013.

3.9.182 ASTM International. ASTM C295, “Standard Guide for Petrographic Examination of Aggregates for Concrete.” West Conshohocken, Pennsylvania: American Society for Testing and Materials. 2012.

3.9.183 ASTM International. ASTM C216, “Standard Specification for Facing Brick (Solid Masonry Units made from Clay or Shale.” West Conshohocken, Pennsylvania: American Society for Testing and Materials. 2016.

3.9.184 ASTM International. ASTM C289, “Standard Test Method for Potential Alkali-Silica Reactivity of Aggregates (Chemical Method).” West Conshohocken, Pennsylvania: American Society for Testing and Materials. 2007.

3.9.185 ASTM International. ASTM C618, “Standard Specification for Coal Fly Ash and Raw or Calcined Natural Pozzolan for Use as a Mineral Admixture in Concrete.” West Conshohocken, Pennsylvania: American Society for Testing and Materials. 1998.

3.9.186 Atkinson, A. and J.A. Hearne. “Mechanistic Model for the Durability of Concrete Barriers Exposed to Sulphate-Bearing Groundwaters.” *Proceedings of the Materials Research Society Conference*. Symposium Proceedings. Pittsburgh, Pennsylvania: Materials Research Society. Vol. 176. pp. 149–156. 1990.

3.9.187 Bastidas-Arteaga, E., M. Sanchez-Silva, A. Chateauneuf, and M. Ribas-Silva. “Coupled Reliability Model of Biodeterioration.” *Chloride Ingress and Cracking for Reinforced Concrete Structures, Structural Safety*. Vol. 30. pp. 110–129. 2008.

3.9.188 Berner, U.R. "Evolution of Pore Water Chemistry During Degradation of Cement in a Radioactive Waste Repository Environment." *Waste Management*. Vol. 12. pp. 201–219. 1992.

3.9.189 Berntz, D.P, M.A. Ehlen, C.F. Ferraris, and E.J. Garboczi. "Sorptivity-Based Service Life Predictions for Concrete Pavements." *7th International Conference on Concrete Pavements, Proceedings*, Vol. 1, Orlando, Florida, September 9–13, 2001. International Society for Concrete Pavements. pp. 181–193. 2001.

3.9.190 Bertolini, L., B. Elsener, P. Pedeferri, and R.P. Polder. *Corrosion of Steel in Concrete: Prevention, Diagnosis, Repair*, 2nd Edition, Wiley-VCH. pp. 409. 2004.

3.9.191 Bouniol, P. and A. Aspart. "Disappearance of Oxygen in Concrete Under Irradiation: The Role of Peroxides in Radiolysis." *Cement and Concrete Research*. Vol. 28. pp. 1,669–1,681. 1998.

3.9.192 Branson, D.E. *Deformation of Concrete Structures*. New York, New York: McGraw-Hill International Book Company. 1977.

3.9.193 Cai, S., L. Cremaschi, and A.J. Ghajar. "Moisture Accumulation and its Impact on the Thermal Performance of Pipe Insulation for Chilled Water Pipes in High Performance Buildings." *International Refrigeration and Air Conditioning Conference at Purdue*, Indiana. July 16–19, 2012. 2012

3.9.194 Cheung, M.M.S., J. Zhao, and Y.B. Chan. "Service Life Prediction of RC Bridge Structures Exposed to Chloride Environments." *Journal of Bridge Engineering*. Vol. 14. pp. 164–178. 2009.

3.9.195 Das, B.J. *Principles of Foundation Engineering*. 4th Edition. Pacific Grove, California: Brooks/Cole Publishing Company. 1999.

3.9.196 Davies, N.F. "Developmental Irradiation Test of SNAP 8 Electrical Components (HF-8), North American Rockwell Corp." NAA-SR- 11924, AT (11- 1)-Gen-8. Canoga Park, California: Atomics International. p. 25. 1966.

3.9.197 Drimalas T., J.C. Clement, K.J. Folliard, R. Dhole, and M.D.A. Thomas. "Laboratory and Field Evaluations of External Sulfate Attack in Concrete." Austin, Texas: Center for Transportation Research, The University of Texas at Austin. 2010.

3.9.198 DuraCrete R17. "Final Technical Report, Probabilistic Performance Based Durability Design of Concrete Structures." BE95-1347/R17. CUR, Gouda, The Netherlands. The European Union–Brite EuRam III. 2000.

3.9.199 EPRI. "Effect of Radiation on Concrete–A Literature Survey and Path Forward." Report 1025584. Palo Alto, California: Electric Power Research Institute. 2012.

3.9.200 Fagerlund, G. "The International Cooperative Test of the Critical Degree of Saturation Method of Assessing the Freeze/Thaw Resistance of Concrete." *Materials and Structures*. Vol. 10. pp. 231–253. 1977.

3.9.201 Fan, J., Z. Luo, and Y. Li. "Heat and Moisture Transfer with Sorption and Condensation in Porous Clothing Assemblies and Numerical Simulation." *International Journal of Heat Mass Transfer*. Vol. 43. pp. 2,989–3,000. 2000.

3.9.202 Figg, J. "ASR-Inside Phenomena and Outside Effects (Crack Origin and Pattern)." Concrete Alkali-Aggregate Reactions. E. Patrick, eds. *Proceedings of the 7th International Conference*. Grattan-Bellew and E. Patrick, eds. Park Ridge, New Jersey: 7th International Conference Organizers. pp. 152–156. 1987.

3.9.203 Freskakis, G.N. "Strength Properties of Concrete at Elevated Temperature." Civil Engineering Nuclear Power. Vol. 1. ASCE National Convention. Boston, Massachusetts: American Society of Civil Engineers. 1979.

3.9.204 Fu, Y. "Delayed ettringite formation in portland cements products." Thesis (Ph.D.). Dept. Civil Engineering. University of Ottawa. Ottawa, Ontario, Canada. 1996.

3.9.205 Gellrich, G. "Calvert Cliffs Nuclear Power Plant." Letter to U.S. Nuclear Regulatory Commission, Response to Request for Supplemental Information. RE: Calvert Cliffs Independent Spent Fuel Storage Installation License Renewal Application (TAC No. L24475). ADAMS Accession No. ML12212A216. 2012.

3.9.206 Ghafoori, N. and R. Mathis. "Sulfate Resistance of Concrete Pavers." *Journal of Materials in Civil Engineering*. Vol. 9. pp. 35–40. 1997.

3.9.207 Giannantonio, D.J., J.C. Kurth, K.E. Kurtis, and P.A. Sobechy. "Effects of Concrete Properties and Nutrients on Fungal Colonization and Fouling." *International Biodeterioration and Biodegradation*. Vol. 63. pp. 252–259. 2009.

3.9.208 Glass, G.K. and N.R. Buenfeld. "The Presentation of the Chloride Threshold Level for Corrosion of Steel in Concrete." *Corrosion Science*. Vol. 39. pp. 1,001–1,013. 1997.

3.9.209 Glauz, D.L., D. Roberts, V. Jain, H. Moussavi, R. Llewellen, and B. Lenz. "Evaluate the Use of Mineral Admixtures in Concrete to Mitigate Alkali-Silica Reactivity." Report FHWA/CA/OR 97-01. Sacramento, California: Office of Materials Engineering and Testing Services. California Department of Transportation. 1996.

3.9.210 Gutt, W.H. and W.H. Harrison. "Chemical Resistance of Concrete." *Concrete*. Vol. 11. pp. 35–37. 1997.

3.9.211 Harmathy, T.Z. "Thermal Properties of Concrete at Elevated Temperatures." *Journal of Materials*. Vol. 5. pp. 47–74. 1970.

3.9.212 Hilsdorf, H.R., J. Kroop, and H.J. Koch. "The Effects of Nuclear Radiation on the Mechanical Properties of Concrete." *Douglas McHenry International Symposium on Concrete and Concrete Structures*. American Concrete Institute Publication SP-55. 1978.

3.9.213 Hobbs, D.W. "Expansion and Cracking in Concrete Associated with Delayed Ettringite Formation." *Ettringite, the Sometimes Host of Destruction*. B. Erlin, ed. SP177 Farmington Hills, Michigan: American Concrete Institute International. pp. 159–181. 1999.

3.9.214 Hu, J., D. Hahn, W. Rudzinski, Z. Wang, and L. Estrada. “Evaluation, Presentation and Repair of Microbial Acid-Produced Attack of Concrete.” Report No. FHWA/TX-11/0-6137-1. Texas Department of Transportation Research and Technology Implementation Office. 2011.

3.9.215 IAEA. “Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings.” IAEA–TECDOC–1025. Vienna, Austria. 1998.

3.9.216 Johansen, V. and N. Thaulow. “Heat Curing and Late Formation of Ettringite.” ACI SP-177. Bernard Erlin, ed. Farmington Hills, Michigan: American Concrete Institute. pp. 199–206. 1999.

3.9.217 Kontani, O., Y. Ichikawa, A. Ishizawa, M. Takizawa, and O. Sato. “Irradiation Effects on Concrete Structures.” *Proceedings of International Symposium on the Ageing Management & Maintenance of Nuclear Power Plants*. pp. 173–182. 2010.

3.9.218 Magniont, C., M. Coutand, A. Bertron, X. Cameleyre, C. Lafforgue, S. Beaufort, and G. Escadeillas. “A New Test Method to Assess the Bacterial Deterioration of Cementitious Materials.” *Cement Concrete Research*. Vol. 41. pp. 429–438. 2011.

3.9.219 Manjeeth K.V. and J.S.K. Rama. “An Experimental Investigation on the Behavior of Portland Cement Concrete and Geopolymer Concrete in Acidic Environment.” *SSRG International Journal of Civil Engineering*. Vol. 2, Issue 5. 2015.

3.9.220 Marchand J., M. Pigeon, D. Bager, and C. Talbot. “Influence of Chloride Solution Concentration of Salt Scaling Deterioration of Concrete.” *ACI Materials Journal.* pp. 429–435. 1999.

3.9.221 Marchand, J., E.J. Sellevold, and M. Pigeon. “Deicer Salt Scaling Deterioration–An Overview.” SP-145. American Concrete Institute. pp. 1–46. 1994.

3.9.222 McDonald, J.E. “An Experimental Study of Multiaxial Creep in Concrete.” American Concrete Institute Special Publication No. 34. Detroit, Michigan: Concrete for Nuclear Reactors. pp. 732–768. 1972.

3.9.223 Mehta, P.K. *Concrete, Structure, Properties and Materials.* Upper Saddle River, New Jersey: Prentice-Hall, Inc. 1986.

3.9.224 Milde, K., W. Sand, W. Wolff, and E. Bock. “Thiobacilli of the Corroded Concrete Walls of the Hamburg Sewer System.” *Journal of General Microbiology*. Vol. 129. pp. 1,327–1,333. 1983.

3.9.225 Mindess S. and J.F. Young. *Concrete*. Englewood Cliffs, New Jersey: Prentice-Hall, Inc. 1981.

3.9.226 Mori, T., T. Nonaka, K. Tazak, M. Koga, Y. Hikosaka, and S. Nota. “Interactions of Nutrients, Moisture, and pH on Microbial Corrosion of Concrete Sewer Pipes. *Water Research*. Vol. 26. pp. 29–37. 1992.

3.9.227 Naus, D.J. “A Review of the Effects of Elevated Temperature on Concrete Materials and Components with Particular Reference to the Modular High-Temperature Gas-Cooled Reactor (MHTGR).” ORNL/NRC/LTR-88/2, LTR Report CTP-88-01. Concrete Technology Program. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1988.

3.9.228 Naus, D.J. “Concrete Properties in Nuclear Environment–A Review of Concrete Material Systems for Application to Pre-Stressed Concrete Pressure Vessels.” ORNL/TM-7632. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1981.

3.9.229 NAVFAC. “Foundations and Earth Structures.” Design Manual NAVFAC DM-7.02. Alexandria, Virginia: U.S. Naval Facilities Engineering Command. 1996.

3.9.230 NAVFAC. “Soil Mechanics.” Design Manual NAVFAC DM-7.01. Alexandria, Virginia: U.S. Naval Facilities Engineering Command. 1986.

3.9.231 Neville, A.M. and W. Dilger. “Creep of Concrete: Plain, Reinforced and Prestressed.” Amsterdam, Holland: North-Holland Publishing Co. 1970.

3.9.232 NRC. “Expert Panel Workshop on Degradation of Concrete in Spent Nuclear Fuel Dry Cask Storage Systems, Official Transcript of Proceedings.” ADAMS Accession Nos. ML15093A003, ML15093A004. Washington, DC: U.S. Nuclear Regulatory Commission. 2015.

3.9.233 NRC. NUREG/CR-6900, “The Effect of Elevated Temperature on Concrete Materials and Structures—A Literature Review.” Washington, DC: U.S. Nuclear Regulatory Commission. 2006.

3.9.234 NRC. NUREG–1557, “Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal.” Washington, DC: U.S. Nuclear Regulatory Commission. 1996.

3.9.235 Okabe, S., O. Mitsunori, I. Tsukasa, and S. Hisashi. “Succession of Sulfur-Oxidizing Bacteria in the Microbial Community on Corroding Concrete in Sewer Systems.” *Applied Environmental Microbiology*. Vol. 73. pp. 971–980. 2007.

3.9.236 Page, C.L. *Nature*. Vol. 297, No. 5,862. pp. 109–115. 1982.

3.9.237 Pavlik, V. “Corrosion of Hardened Cement Paste by Acetic and Nitric Acids: Part I. Calculation of Corrosion Depth.” *Cement and Concrete Research*. Vol. 24. pp. 551–562. 1994.

3.9.238 Pavlik, V. and S. Uncik. “The Rate of Corrosion of Hardened Cement Pastes and Mortars with Additive of Silica Fume in Acids.” *Cement and Concrete Research*. Vol. 27. pp. 1,731–1,745. 1997.

3.9.239 Pedneault, A. “Development of testing and analytical procedures for the evaluation of the residual potential of reaction, expansion, and deterioration of concrete affected by ASR.” M.Sc. Memoir. Laval University. Québec City, Canada. pp. 133. 1996.

3.9.240 Perenchio, W.F., I. Kaufman, and R. J. Krause. “Concrete Repair in a Desert Environment.” *Concrete International*. Vol. 13, No. 2. Farmington Hills, Michigan: American Concrete Institute. pp. 23–25. 1991.

3.9.241 Perez, M., M. Garcia, L. Transversa and M. Stupak. “Concrete Deterioration by Golden Mussels.” *Proceedings of International RILEM Conference on Microbial Impact on Building Materials*. M. Ribas Silva ed. Lisbon, Portugal. pp. 39-47. 2003.

3.9.242 Phan, L.T. and N.J. Carino. “Fire Performance of High Strength Concrete: Research Needs.” *Advanced Technology in Structural Engineering. ASCE/SEI Structures Congress 2000.* Proceedings. Philadelphia, Pennsylvania. 2000.

3.9.243 Pigeon, M. “Frost Resistance, A Critical Look.” Concrete Technology, Past, Present, and Future. *Proceedings of V. Mohan Malhotra Symposium*. American Concrete Institute. SP-144. pp. 141–158. 1994.

3.9.244 Poe, W.L. “Final Long-Term Degradation of Concrete Facilities Presently Used for Storage of Spent Nuclear Fuel and High-Level Waste.” Rev. 1. Tetra Tech NUS, Inc. Aiken, South Carolina: Degradation Mechanisms for Concrete and Reinforcing Steel. 1998.

3.9.245 Poole, A.B. “Introduction to Alkali-Aggregate Reaction in Concrete.” R.N. Swamy and R. Van Nostrand, eds. New York, New York: *The Alkali-Silica Reaction in Concrete*. 1992.

3.9.246 Sahu, S. and N. Thaulow “Delayed Ettringite Formation in Swedish Concrete Railroad Ties.” *Cement and Concrete Research*. Vol. 34. pp. 1,675–1,681. 2004.

3.9.247 Sanchez-Silva, M. and D. Rosowsky. “Biodeterioration of Construction Materials: State of the Art and Future Challenges.” *Journal of Materials in Civil Engineering*. Vol. 20. pp. 352–365. 2008.

3.9.248 Sawan, J. “Cracking Due to Frost Action in Portland Cement Concrete Pavements–A Literature Survey, Concrete Durability.” *Proceedings of Katharine and Bryant Mather International Conference*. American Concrete Institute. SP-100. pp. 781–802. 1987.

3.9.249 Schiessl, P., P. Bamforth, V. Baroghel-Bouny, G. Corley, M. Faber, J. Forbes, C. Gehlen, P. Helene, S. Helland, T. Ishida, G. Markeset, L. Nilsson, S. Rostam, A.J.M. Siemes, and J. Walraven. “Model Code for Service Life Design.” Lausanne, Switzerland: Fib Bulletin No. 34. 2006

3.9.250 Schneider, U., U. Diederichs, and C. Ehm. “Effect of Temperature on Steel and Concrete for PCRV’s.” *Nuclear Engineering and Design*. Vol. 67. pp. 245–258. 1981.

3.9.251 Shayan, A. and G.W. Quick. “Microscopic Features of Cracked and Uncracked Concrete Railway Sleepers.” *ACI Materials Journal*. Vol. 89. pp. 348–361. 1992.

3.9.252 Snead, L.L., S.J. Zinkle, and D.P. White. “Thermal Conductivity Degradation of Ceramic Materials Due to Low Temperature, Low Dose Neutron Irradiation.” *Journal of Nuclear Materials*. Vol. 340. pp. 187–202. 2005.

3.9.253 Snead, L.L., D. Steiner, and S.J. Zinkle. "Measurement of the Effect of Radiation Damage to Ceramic Composite Interfacial Strength." *Journal of Nuclear Materials*. Vol. 191–194. 14 pp. 566–570. 1992.

3.9.254 Snead, L.L., R. Yamada, K. Noda, Y. Katoh, S.J. Zinkle, W.S. Eatherly, and A.L. Qualls. "In Situ Thermal Conductivity Measurement of Ceramics in a Fast Neutron Environment." *Journal of Nuclear Materials*. Vol. 283–287. pp. 545–550. 2000.

3.9.255 Stark, D. "The Moisture Condition of Field Concrete Exhibiting Alkali-Silica Reactivity." *CANMET/ACI Second International Conference on Durability of Concrete*. SP-126. Farmington Hills, Michigan. American Concrete Institute. pp. 973–987. 1991.

3.9.256 Szklarska-Smialowska, Z. *Pitting Corrosion of Metals*. Houston, Texas: National Association of Corrosion Engineers. 1986.

3.9.257 Tang, L. and P. Sandberg. "Chloride Penetration into Concrete Exposed Under Different Conditions." *Durability of Building Materials and Components 7*. Vol. 1. C. Sjöström, eds. Stockholm, Sweden. 1996.

3.9.258 Thomas, M.D.A., B. Fournier, and K.J. Folliard. *Alkali-Aggregate Reactivity (AAR) Facts Book*. Austin, Texas: The Transtec Group, Inc. 2013.

3.9.259 Thomas, M., K. Folliard, T. Drimalas, and T. Ramlochan "Diagnosing Delayed Ettringite Formation in Concrete Structures." *Cement and Concrete Research*. Vol. 38. pp. 841–847. 2008.

3.9.260 Trejo, D., P.D. Figueiredo, M. Sanchez, C. Gonzalez, S. Wei, and L. Li. "Analysis and Assessment of Microbial Biofilm-Mediated Concrete Deterioration." Texas Transportation System. Texas Transportation System. The Texas A&M University System. 2008.

3.9.261 Ueda, H., Y. Kimachi, S. Ushijima, and K. Shyuttoh. "Deterioration Model of Acid-Rain-Affected Concrete and Test Results of Ordinary and Super Quality Concrete." *26th Conference on Our World in Concrete & Structures*. Singapore. 2001.

3.9.262 U.S. Department of the Army. "Engineering and Design: Settlement Analysis." EM 1110-1-1904. Washington, DC: U.S. Army Corps of Engineers. September 30, 1990.

3.9.263 Vafai, K. and S. Sarkar. "Condensation Effects in a Fibrous Insulation Slab. *Journal of Heat Transfer*. Vol. 108, No. 8. pp. 667–675. 1986.

3.9.264 Van Dam, T. and D. Peshkin. "Concrete Aggregate Durability Study." Final Report 5756. Urbana, Illinois: Applied Pavement Technology, Inc. 2009.

3.9.265 Verbeck, C.J. and P. Klieger. "Studies of Salt Scaling of Concrete." *Highway Research Bulletin*. 6 Vol. 150. pp. 1–17. 1957.

3.9.266 Vollertsen, J., A.H. Nielsen, H.S. Jensen, W.A. Tove, and H.J. Thorkild. "Corrosion of Concrete Sewers—The Kinetics of Hydrogen Sulfide Oxidation." *Science of the Total Environment*. Vol. 394. pp. 162–170. 2008.

3.9.267 Wang, C.K and C.G. Salmon. *Reinforced Concrete Design*. 6th Edition. New York, New York: Addison-Wesley. 1998.

3.9.268 Webster, R.P. and L.E. Kukacka. “Effects of Acid Deposition on Portland Cement Concrete.” *Materials Degradation Caused by Acid Rain*. ACS Symposium Series. Vol. 318. pp. 239–249. 2009.

3.9.269 Wei, S., Z. Jiang, H. Liu., D. Zhou, and M. Sanchez-Silva. “Microbiologically Induced Deterioration of Concrete—A Review.” *Brazilian Journal of Microbiology*. Vol. 44. pp. 1,001–1,007. 2013.

3.9.270 Weiss, C. A., Jr., M.C. Sykes, T.S. Poole, J.G. Tom, B.H. Green, B.D. Neeley, and P.G. Malone. “Controlling Sulfate Attack in Mississippi Department of Transportation Structures.” Vicksburg, Mississippi: U.S. Army Engineer Research and Development Center. 2009.

3.9.271 Xu, H. “On the Alkali Content of Cement in AAR.” *Concrete Alkali Aggregate Reactions. Proceedings of the 7th International Conference*. Grattan-Bellew and E. Patrick, eds. Park Ridge, New Jersey: 7th International Conference Organizers. pp. 451–455. 1987.

3.9.272 Yano, T., K. Ichikawa, M. Akiyoshi, and Y. Tachi. “Neutron Irradiation Damage in Aluminum Oxide and Nitride Ceramics Up to a Fluence of 4.2 × 1026 n/m2” *Journal of Nuclear Materials*. Vol. 283–287. pp. 947–951. 2000.

3.9.273 Adamson, R., B. Cox, J. Davies, P. Rudling, S. Vidyanathan. “IZNA-6 Special Topical Report: Pellet-Cladding Interaction (PCI and PCMI),” R. Adamson, ed. Skultuna, Sweden: Advanced Nuclear Technology International. 2006.

3.9.274 Ahn, T., V. Rondinella, and T. Wiss. “Potential Stress on Cladding Imposed by the Matrix Swelling from Alpha Decay in High Burnup Spent Nuclear Fuel.” Paper 6830. *2013 International High-Level Radioactive Waste Management Conference*, April 28–May 2. Albuquerque, New Mexico: American Nuclear Society. 2013.

3.9.275 Ankem, R. and T. Wilt. “A Literature Review of Low Temperature (< 0.25 Tmp) Creep Behavior α, α-β, and β Titanium Alloys.” ADAMS Accession No. ML072060401. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2006.

3.9.276 Bai, J., J. Gilbon, C. Prioul, and D. Francois. “Hydride Embrittlement in Zircaloy-4 Plate, Part I, Influence of Microstructure on the Hydride Embrittlement in Zircaloy-4 at 20°C and 350°C”and Part II, “Interaction Between the Tensile Stress and the Hydride Morphology.” *Metallurgical and Materials Transactions A*. Vol. 25A, Issue 6. pp. 1,185–1,197. June 1994.

3.9.277 Billone, M.C., T.A. Burtseva, and Y.Y. Liu. “Characterization and Effects of Hydrides in High-Burnup PWR Cladding Alloys.” *Proceedings of the International High-Level Radioactive Waste Management Conference*, Charleston, South Carolina. Paper No. 12617. American Nuclear Society. April 12–16, 2015.

3.9.278 Bossis, P., B. Verhaeghe, S. Doriot, D. Gilbon, V. Chabretou, A. Dalmais, J.P. Mardon, M. Blat and A. Miquet. "In PWR Comprehensive Study of High Burn-up Corrosion and Growth Behaviour of M5 and Recrystallised Low-Tin Zircaloy-4." *15th ASTM International Symposium: Zirconium in the Nuclear Industry*. Sun River, Oregon. ASTM International. June 20, 2007.

3.9.279 Cappelaere, C., R. Limon, T. Bredel, P. Herter, D. Gilbon, S. Allegre, P. Bouffioux and J.P. Mardon. "Long Term Behaviour of the Spent Fuel Cladding in Dry Storage Conditions." *8th International Conference on Radioactive Waste Management and Environmental Remediation.* October 2001. Vol. 2. Bruges, Belgium. American Society of Mechanical Engineers. 2001.

3.9.280 Cazalis, B., C. Bernaudat, P. Yvon, J. Desquines, C. Poussard, and X. Averty. "The PROMETRA program: A Reliable Material Database for Highly Irradiated Zircaloy-4, ZIRLO™ and M5™ fuel claddings." Proceeding of the 18th International Conference on Structural Mechanics in Reactor Technology. 18th ed., Paper SMiRT18-C02-1. August 2005.

3.9.281 Chan, K.S. "A Micromechanical Model for Predicting Hydride Embrittlement in Nuclear Fuel Cladding Material." *Journal of Nuclear Materials.* Vol. 227. pp. 220–236. 1996.

3.9.282 Chan, K.S. "An Assessment of Delayed Hydride Cracking in Zirconium Alloy Cladding Tubes Under Stress Transients." *International Materials Reviews*. Vol. 58, No. 6. pp. 349–373. 2013.

3.9.283 Coleman, C., V. Grigoriev, V. Inozemtsev, V. Markelov, M. Roth, V. Makaevicius, Y.S. Kim, K.L. Ali, J.K. Chakravartty, R. Mizrahi, and R. Lalgud. "Delayed Hydride Cracking in Zircaloy Fuel Cladding—An IAEA Coordinated Research Programme." *Nuclear Engineering and Technology.* Vol. 41, No. 2. pp. 171–177. 2009.

3.9.284 Cox, B. "Hydrogen Trapping by Oxygen and Dislocations in Zirconium Alloys." *Journal of Alloys and Compositions*. Vol. 256 pp. L4–L7. 1997.

3.9.285 Cox, B. "Degradation of Zirconium Alloys in Water Cooled Reactors." *Proceedings of the Third International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors*, Warrendale, Pennsylvania: The Metallurgical Society. pp. 65–76. 1988.

3.9.286 Cox, B. "Oxidation of Zirconium and its Alloys." *Advances in Corrosion Science and Technology*. M. Fontana and R.W. Staehle, eds. New York, New York: Plenum Press. 1976.

3.9.287 Crescimanno, P.J., W.R. Campbell, and I. Goldberg. "A Fracture Mechanics Mode for Iodine Stress Corrosion Crack Propagation in Zircaloy Tubing." In Environment-Sensitive Fracture Evaluation and Comparison of Test Methods. ASTM STP 821 (S.W. Dean, E.N. Pugh and O.M. Ugiansky, eds). Philadelphia, Pennsylvania: American Society for Testing and Materials. pp.150–169. 1984.

3.9.288 Daum, R.S., S. Majumdar, Y. Liu, and M.C. Billone. "Radial-hydride Embrittlement of High-Burnup Zircaloy-4 Fuel Cladding." *Journal of Nuclear Science and Technology*. Vol. 43, No. 9. pp. 1,054–1,067. 2006.

3.9.289 Devoe, R. and K.R. Robb. “COBRA-SFS Dry Cask Modeling Sensitivities in High-Capacity Canisters.” *Proceedings of the International High-Level Radioactive Waste Management Conference*, April 12–16, 2015. Paper No. 12701. Charleston, South Carolina. 2015.

3.9.290 EPRI. “High Burnup Dry Storage Cask Research and Development Project: Final Test Plan.” DE-NE-0000593. Palo Alto, California: Electric Power Research Institute. 2014.

3.9.291 EPRI. “Extended Storage Collaboration Program (ESCP) Progress Report and Review of Gap Analyses.” Report 1022914. Palo Alto, California: Electric Power Research Institute. 2011.

3.9.292 EPRI. “Temperature Limit Determination of the Inert Dry Storage of Spent Nuclear Fuel.” Report TR-103949. Palo Alto, California: Electric Power Research Institute. 1997.

3.9.293 Foregeaud, S., J. Desquines, M. Petit, C. Getrey, and G. Sert. “Mechanical Characteristics of Fuel Rod Claddings in Transport Conditions,” *Packaging, Transport, Storage, & Security of Radioactive Material*. Vol. 20. pp. 69–76. 2009.

3.9.294 Fuketa, T., T. Sugiyama, T. Nakamura, H. Sasajima, and F. Nagase. NUREG/CP-01 85, “Effects of Pellet Expansion and Cladding Hydrides on PCMI Failure of High Burnup LWR Fuel During Reactivity Transients.” Nuclear Safety Research Conference. Washington, DC. 2003.

3.9.295 Gilbert, E.R., E.P. Simonen, C.E. Beyer, and P.G. Medvedev. “Update of CSFM Methodology for Determining Temperature Limits for Spent Fuel Dry Storage in Inert Gas.” ADAMS Accession No. ML022250067. Washington, DC: U.S. Nuclear Regulatory Commission. 2001.

3.9.296 Geelhood, K.J., C.E Beyer, and W.G Luscher. “PNNL Stress/Strain Correlation for Zircaloy.” Pacific Northwest National Laboratory. PNNL-17700. July 2008.

3.9.297 Geelhood, K.J. and W.G. Luscher. “FRAPCON-3.5: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup.” Pacific Northwest National Laboratory. PNNL-19418 Vol. 1. Rev. 1. NUREG/CR-7022. Vol. 1, Rev. 1. ADAMS Accession No. ML14295A539. October 2014.

3.9.298 Green, D. W. and R. Perry. *Perry’s Chemical Engineers’ Handbook. Eighth Edition.* McGraw-Hill Education, New York, New York, 2007.

3.9.299 Hanson, B. D. “High Burnup Fuel, Associated Data Gaps, and Integrated Approach for Addressing the Gaps,” Presented to Nuclear Waste Technical Review Board. <http://www.nwtrb.gov/meetings/2016/feb/hanson.pdf> February 29, 2016.

3.9.300 Haynes, W.M., D.R. Lide, and T.J. Bruno. *CRC Handbook of Chemistry and Physics. 93rd Edition*. CRC Press. Boca Raton, Florida. 2013.

3.9.301 International Atomic Energy Agency (IAEA). "Corrosion of zirconium alloys in nuclear power plants." Vienna, Austria: TECDOC-684. January 1993.

3.9.302 Ibarra, L., T. Wilt, G. Ofoegbu, and A. Chowdhury. "Structural Performance of Drip Shield Subjected to Static and Dynamic Loading." ADAMS Accession No. ML070240131. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2007.

3.9.303 INCO. "Corrosion Resistance of the Austenitic Chromium-Nickel Stainless Steels in Atmospheric Environments." The International Nickel Company (INCO), Inc. Suffern, New York. 1970. <http://www.ohiogratings.com/pdfs/StainlessSteelCorrosionStudy.pdf>

3.9.304 Ito, K., K. Kamimura, and Y. Tsukuda. "Evaluation of Irradiation Effect on Spent Fuel Cladding Creep Properties." *2004 International Meeting on LWR Fuel Performance*, Orlando, Florida. September 19–22, 2004. American Nuclear Society. p. 440. 2004.

3.9.305 Jaworski, A. and S. Ankem. "Influence of the Second Phase on the Room-Temperature Tensile and Creep Deformation Mechanisms of α-β Titanium Alloys: Part I. Tensile Deformation." *Metallurgical and Materials Transactions*. Vol. 37A. pp. 2,739–2,754. 2006.

3.9.306 Jernkvist, L.O., A. R. Massih, and P. Rudling. "A Strain-Based Clad Failure Criterion for Reactivity Initiated Accidents in Light Water Reactors." SKI Report 2004:32. Uppsala, Sweden: 2004.

3.9.307 Jung, H., P. Shukla, T. Ahn, L. Tipton, K. Das, X. He, and D. Basu. "Extended Storage and Transportation: Evaluation of Drying Adequacy." ADAMS Accession No. ML13169A039. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2013.

3.9.308 Kain, V. "Chapter 5: Stress Corrosion Cracking in Stainless Steels." In *Stress Corrosion Cracking: Theory and Practice*. V.S. Raja and T. Shoji, eds. Cambridge, England: Woodhead Publishing. pp. 199–244. 2011.

3.9.309 Kim, Y.S. "Kinetics of Crack Growth in Zirconium Alloys (I): Temperature Dependence of the Crack Growth Rate." *Journal of Applied Physics.* Vol. 106. pp. 123,520-1–123,520-6. 2009.

3.9.310 Kim, Y.S. "Hydride Reorientation and Delayed Hydride Cracking of Spent Fuel Rods in Dry Storage." *Metallurgical and Materials Transactions A*. Vol. 40A. pp. 2,867–2,875. 2009.

3.9.311 Kim, Y.S. "Delayed Hydride Cracking of Spent Fuel Rods in Dry Storage." *Journal of Nuclear Materials*. Vol. 378. pp. 30–34. 2008.

3.9.312 Kim, J.-S., Y.-J. Kim, D.-H. Kook, and Y.-S. Kim. "A Study on Hydride Reorientation of Zircaloy-4 Cladding Tube Under Stress," *Journal of Nuclear Materials*, Vol. I 456, pp. 246–252, 2015.

3.9.313 Kim, J.-S., T.-H. Kim, D.-H. Kook, and Y.-S. Kim. "Effects of Hydride Morphology on the Embrittlement of Zircaloy-4 Cladding," *Journal of Nuclear Materials*, Vol. 456, pp. 235–245. 2015.

3.9.314 King, S., R. Kesterson, K. Yueh, R. Comstock, W. Herwig, and S. Ferguson. "Impact of Hydrogen on the Dimensional Stability of ZIRLO Fuel Assemblies." In *Zirconium in the Nuclear Industry: Thirteenth International Symposium*, ASTM STP 1423. West Conshohocken, Pennsylvania: ASTM International. pp. 471-479. 2002.

3.9.315 Kubo, T., Y. Kobayashi, and H. Uchikoshi. "Measurements of Delayed Hydride Cracking Propagation Rate in the Radial Direction of Zircaloy-2 Cladding Tubes." *Journal of Nuclear Materials.* Vol. 427. pp. 18–29. 2012.

3.9.316 Kreyns, P.H., G.L. Spahr, and J.E. McCauley. "An Analysis of Iodine Stress Corrosion Cracking of Zircaloy-4 Tubing." *Journal of Nuclear Materials.* Vol. 61. pp. 203–212. 1976.

3.9.317 Lin, X. and G. Haicheng. "High Cycle Fatigue Properties and Microstructure of Zirconium and Zircaloy-4 Under Reversal Bending." *Materials Science and Engineering A*. Vol. 252. 14 pp. 166–173. 1998.

3.9.318 Luscher, W.G and K.J. Geelhood. "Material Property Correlations: Comparisons Between FRAPCON-3.4, FRAPTRAN 1.4, and MATPRO." PNNL-19417 (NUREG/CR-7024). Richland, Washington: Pacific Northwest National Laboratory. 2010.

3.9.319 Mattas, R.F., F.L. Yagee and L.A. Neimark. "Effect of Zirconium Oxide on the Stress Corrosion Susceptibility of Irradiated Zircaloy Cladding." In *Zirconium in the Nuclear Industry: Fifth International Symposium.* ASTM STP 754, (D.G Franklin, ed. West Conshohocken, Pennsylvania: American Society for Testing and Materials. pp. 158–170. 1982.

3.9.320 Mardon, J. P., G.L. Garner, and P.B. Hoffmann. "M5® A Breakthrough in Zr Alloy." *Proceedings of 2010 LWR Fuel Performance/TopFuel/WRFPM*, Orlando, Florida, September 26–29, 2010. American Nuclear Society. 2010.

3.9.321 Masafumi, N., K. Uchida, A. Miyazaki, and Y. Ishii. "Annealing Study on Neutron Irradiation Effects in Resonance Frequencies of Zircaloy Plates by EMAR Method." *Journal of Nuclear Science and Technology*, Vol. 44, No. 10. pp. 1,285–1,294. 2007.

3.9.322 Morize P., J. Baicry, and J. P. Mardon. "Effect of Irradiation at 588 K on Mechanical Properties and Deformation Behavior of Zirconium Alloy Strip." *Zirconium in the Nuclear Industry: Seventh International Symposium*. ASTM STP 939. R.B. Adamson and L.F.P. Van Swam, eds. ASTM. pp. 101–119. 1987.

3.9.323 Murty, K.L. "The Internal Pressurization Creep of Zr Alloys for Spent-Fuel Dry Storage Feasibility." *Journal of the Minerals, Metals and Materials Society*. Vol. 52, No. 9. pp. 34–43. 2000.

3.9.324 NRC. "Acceptable Fuel Cladding Hydrogen Uptake Models." ADAMS Accession No. ML15133A306. Washington, DC: U.S. Nuclear Regulatory Commission. 2015.

3.9.325 NRC. Draft Regulatory Issue Summary 2015-XXX, "Considerations in Licensing High Burnup Spent Fuel in Dry Storage and Transportation." ADAMS Accession No. ML14175A203. Washington, D.C: U.S. Nuclear Regulatory Commission. 2015.

3.9.326 NRC. Interim Staff Guidance-1, "Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function." Washington, DC: U.S. Nuclear Regulatory Commission. 2007.

3.9.327 NRC. "Safety Evaluation Report Related to the Topical Report for Castor V/21 Dry Spent Fuel Storage Cask Submitted by General Nuclear Systems, Inc." NRC-SER-85-9. Washington, DC: U.S. Nuclear Regulatory Commission. 1985.

3.9.328 Palit, G.C. and H.S. Gadiyar. "Pitting Corrosion of Zirconium in Chloride Solution." *CORROSION*. Vol. 43, No. 3. pp. 140–148. 1987.

3.9.329 Raynaud, P.A.C. and R.E. Einziger. "Cladding Stress During Extended Storage of High Burnup Spent Nuclear Fuel." *Journal of Nuclear Materials*. Vol. 464. pp. 304–312. 2015.

3.9.330 Rebak, R.B. "Chapter 7: Stress Corrosion Cracking (SCC) of Nickel-Based Alloys." In *Stress Corrosion Cracking: Theory and Practice*. V.S. Raja and T. Shoji, eds. Cambridge, England: Woodhead Publishing. pp. 273–306. 2011.

3.9.331 Rondinella, V.V., T. Wiss, E. Maugeri, J.Y. Colle, D. Wegen, and D. Papaioannou. "Effects of He Build-up on Nuclear Fuel Evolution during Storage." *International Workshop on Spent Fuel Integrity in Dry Storage.* Korea Atomic Energy Research Institute. Korea. November 4–5, 2010.

3.9.332 Rondinella, V.V and T. Wiss. "The High Burnup Structure in Nuclear Fuel," *Materials Today*, Vol. 13, pp. 24–32, 2010.

3.9.333 Rondinella, V.V., T. Wiss, D. Papaioannou, and R. Nasyrow. "Studies on Nuclear Fuel Evolution during Storage and Testing of Used Fuel Response to Impact Loadings." PSAM11 ESREL2012. Helsinki, June 25–29, 2012.

3.9.334 Rothman, A.J. "Potential Corrosion and Degradation Mechanisms of Zircaloy Cladding on Spent Nuclear in a Tuff Repository." UCID-20172. Livermore, California: Lawrence Livermore National Laboratory. 1984.

3.9.335 Sasahara, A. and T. Matsumura. "Post-Irradiation Examinations Focused on Fuel Integrity of Spent BWR-MOX and PWR-UO2 Fuels Stored for 20 Years." *Nuclear Engineering and Design.* Vol. 238. pp. 1,250–1,259. 2008.

3.9.336 Scaglione, J.M., G. Radulescu, W.J. Marshall, and K.R. Robb. "A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages." NUREG/CR-7203, ORNL/TM-2013/92. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 2015.

3.9.337 Shimada, S. and M. Nagai. "A Fractographic Study of Iodine-Induced Stress Corrosion Cracking in Irradiated Zircaloy-2 Cladding." *Journal of Nuclear Materials.* Vol. 114. pp. 222–230. 1983.

3.9.338 Shukla, P.K., R. Pabalan, T. Ahn, L. Yang, X. He, and H. Jung. “Cathodic Capacity of Alloy 22 in the Potential Yucca Mountain Repository Environment.” *Proceedings of the CORROSION 2008 Conference, Corrosion in Nuclear Systems Symposium*, New Orleans, Louisiana, March 16–20, 2008. Paper No. 08583. Houston, Texas: NACE International. 2008.

3.9.339 Sidky, P.S. “Iodine Stress Corrosion Cracking of Zircaloy Reactor Cladding: Iodine Chemistry (A Review),” *Journal of Nuclear Materials*, Vol. 256, pp. 1–17. 1998.

3.9.340 Simpson, C.J. and C.E. Ells. “Delayed Hydrogen Embrittlement in Zr-2.5 wt % Nb.” *Journal of Nuclear Materials*. Vol. 52. pp. 289–295. 1974.

3.9.341 Thomazet, J. et al. “The Corrosion of the Alloy M5™: An Overview.” IAEA Technical Committee Meeting on Behavior of High Corrosion Zr-Based Alloys. Buenos Aires, Argentina: October 24–28, 2005.

3.9.342 Torimaru, T., T. Yasuda, and M. Nakatsuka. “Changes in Mechanical Properties of Irradiated Zircaloy-2 Fuel Cladding Due to Short-Term Annealing.” *Journal of Nuclear Materials*. Vol. 238. pp. 169–174. 1996.

3.9.343 Tsai, H. and M.C. Billone. NUREG/CP-0180, “Characterization of High-Burnup PWR and BWR Rods, and PWR Rods After Extended Dry-Cask Storage.” Proceedings of the 2002 Nuclear Safety Research Conference, October 28–30, 2002. pp. 157–168. Washington, DC: U.S. Nuclear Regulatory Commission. 2003.

3.9.344 Van Rooyen, D. and H.R. Copson. “Metal Corrosion in the Atmosphere.” Report No. STP 435. West Conshohocken, Pennsylvania: ASTM International. 1968.

3.9.345 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.

3.9.346 Wang, J. J.-A. “Cyclic Integrated Reversible-bending Fatigue Tester (CIRFT) Framework Approaches and Analytical Evaluations.” Oak Ridge National Laboratory (ORNL). Presented at Extended Storage Collaboration Program (ESCP) Meeting. Electric Power Research Institute (EPRI). Charlotte, North Carolina, December 2–4, 2014.

3.9.347 Wang, J. J.-A. ORNL, 2014 ASTM C26 Committee Meeting, June, 2014.

3.9.348 Wisner, S. and R. Adamson. “Combined Effects of Radiation Damage and Hydrides on the Ductility of Zircaloy-2.” *Nuclear Engineering and Design*. Vol. 185. pp. 33–49. 1998.

3.9.349 Wisner, S.B. and R.B. Adamson. “Embrittlement of Irradiated Zircaloy by Cadmium and Iodine.” *Embrittlement by Liquid and Solid Metals*. M.H. Kamdar, ed. Metallurgical Society of AIME. pp. 437–456. 1982.

3.9.350 Yagee, F.L., R.F. Mattas, and L.A. Neimark. “Characterization of Irradiated Zircaloys: Susceptibility to Stress Corrosion Cracking.” Interim Report. EPRI NP-1557. Palo Alto, California: Electric Power Research Institute. October 1980.

3.9.351 EPRI. “Characterization of Irradiated Zircaloys: Susceptibility to Stress Corrosion Cracking.” Interim Report, EPRI NP-1155. Palo Alto, California: Electric Power Research Institute. September 1979.

3.9.352 [DELETED]

3.9.353 EPRI Report No. TR-106440, “Evaluation of Expected Behavior of LWR Stainless Steel-Clad Duel in Long-Term storage”

3.9.354 Bauer, W., and W.D. Wilson. “Helium Mitigation in Metals.” Radiation-Induced Voids in Metals, CONF-710601, pp.230-247. Proceedings of the 1971 International Conference held at Albany, New York, U.S. Atomic Energy Commission, June 1971

3.9.355 Johnson, A.B., Jr. *Behavior of Spent Nuclear Fuel in Water Pool Storage.* BNWL-2256, Pacific Northwest Laboratory, Richland Washington, 1977.

3.9.356 Johnson, A.B., Jr., E.R. Gilbert, D.R. Oden, D.L. Weeks, and J.C. Dobbins, “Simulated Dry Storage Test of a Spent PWR Nuclear Fuel Assembly in Air.” Waste Management, Tucson Arizona, 1985.

3.9.357 McKinnon, M.A., and V.A. Deloach, “Spent Nuclear Fuel Storage – Performance Tests and Demonstrations, PNL-8451, Pacific Northwest Laboratory, Richland Washington, 1993.

3.9.358 Miller, D.A., T.G. Langdon, “Creep Fracture Maps for 316 Stainless Steel”, Met. Trans. A, Vol. 10A, pp. 1635-1641. American Society of Metals and The Metallurgical Society of AIME, 1979.

3.9.359 Forcey. K.S., D.K. Ross, J.C.B. Simpson, and D.S. Evans. “Hydrogen Transportation and Solubility in 316L and 1.4914 Steels for Fusion Reactor Applications.” Journal of Nuclear Materials, 1988.

3.9.360 Cunningham, M.E., E.P. Simonem, R.T. Allemann, I.S. Levy, R.F. Hazelton, and E.R. Gilbert. “Control of Degradation of Spent Fuel During Dry Storage in an Inert Atmosphere”, PNL-6364, Pacific Northwest Laboratory, Richland Washington, 1987.

3.9.361 Nelson, H.G. “Testing for Hydrogen Environment Embrittlement: Primary and Secondary Influences”, Hydrogen Embrittlement Testing, pp 153-169, ASTM STP 543, American Society for Testing and Materials, Philadelphia Pennsylvania 1974.

3.9.362 Gilbert, E.R., W.J. Bailey, A.B. Johnson, Jr., and McKinnon, “Advances in Technology for Storing Light Water Reactor Spent Fuel.” Nuclear Technology, 89(1990): 141-161, 1990.

3.9.363 Johnson, A.B., Jr. “A Review of Corrosion Phenomena on Zirconium Alloys, Niobium, Titanium, Inconel, Stainless Steel, and Nickel Plate Under Irradiation.” Reviews on Coatings and Corrosion, Vol. I, No. 4, Fruend Publishing House Ltd., Tel-Aviv. 1975.

3.9.364 McElory, W.N., and H. Farrar IV. “Helium Production in Stainless Steel and its Constituents as Related to LMFBR Development Programs.” Radiation-Induced Voids in Metals. U.S. Atomic Energy Commission, Washington, D.C.” 1971.

3.9.365 NAC Technical Report ED20170046, “NAC-UMS and NAC-MPC ISFSI and Individual TSC Rankings Based on EPRI CISCC Criteria,” dated April 18, 2017.

3.9.366 NAC Calculation 30013-2004, Revision 0, Evaluation of Stainless Steel Clad Spent Fuel for Fatigue in Storage.

3.9.367 NRC. “Study of ACI 349.3R-02 Tier 2 (i.e., Section 5.2.1) Criteria Impacts on Dose Rates for Several Spent Nuclear Fuel Storage System Designs”, May 2019

3.9.368 ASME Code Case N-860, “Inspection Requirements and Evaluation Standards for Spent Nuclear Fuel Storage and Transport Containment Systems,” July 2020.

3.9.369 ASME Code, Section XI, Division 1, Subsection IWE, 2007

APPENDIX

APPENDIX

Appendix A

Aging Management Program
NAC-MPC CoC 72-1025

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Table A-1
AMP-1 - Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSC)

AMP Element	AMP Description
1. Program Scope	Examination of welded stainless-steel dry storage Transportable Storage Canisters (TSC) readily accessible (1) external surfaces for localized corrosion and stress corrosion cracking (SCC). (1) The accessible surfaces of the TSC are defined as those surfaces that can be examined using a given examination method without moving the TSC.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions.
3. Parameters Monitored/ Inspected	Parameters monitored and/or inspected include: • Visual evidence of localized corrosion, including pitting corrosion and crevice corrosion, and SCC. • Size and location of localized corrosion and SCC on TSC welds and heat affected zones (HAZs) (≤ 2 inches [50mm] from weld edge). • Appearance and location of discontinuities on the examined TSC surfaces.
4. Detection of Aging Effects	Method or Technique Aging effects are detected and characterized by: • General visual examination using direct or remote methods of the TSC accessible external surfaces away from the weld region for localized corrosion and anomalies. • Visual screening examination by direct or remote means of accessible TSC welds, associated HAZs, and known areas of removed temporary attachments and weld repairs using qualified VT-3 methods and equipment to identify corrosion products that may be indicators of localized corrosion and SCC. • An assessment examination meeting the requirements of VT-1 is required if the screening examination identifies any visual anomaly that is not consistent with prior results or is identified for the first time. • A supplemental examination is required for any visual anomaly within the weld region that is classified as a major indication as discussed in Section 6, Acceptance Criteria. • The extent of coverage shall be maximized subject to the limits of accessibility. Sample Size For sites conducting a TSC examination there should be a minimum of one TSC examined at each site. Preference should be given to the TSC(s) with the greatest susceptibility for localized corrosion or SCC. Frequency • Baseline inspection at beginning of the period of extended operation • Every 10 years for TSCs without detection of indications of major corrosion degradation or SCC • Every 5 years for TSCs with detection of major indications of corrosion degradation or detection(s) of SCC

Table A-1
AMP-1 - Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSC) (continued)

AMP Element	AMP Description
4. Detection of Aging Effects (continued)	Data Collection Documentation of the examination of the TSC, location and appearance of deposits, and an assessment of the suspect areas where corrosion products and/or SCC were observed as described in corrective actions shall be maintained in the licensee's record retention system. Timing of Inspections The baseline inspection shall be performed within 1-year after the 20th anniversary of the first cask loaded at the ISFSI, or within 1-year after the effective date of the CoC renewal if CoC is in period of timely renewal, whichever is later, whichever is later unless otherwise justified.
5. Monitoring and Trending	Monitoring and trending methods will: • Establish a baseline at the beginning of the period of extended operation for the selected TSC. • Track and trend on subsequent inspections of the selected TSC: o The appearance of the selected TSC, particularly at welds and crevice locations documented with images and/or video that will allow comparison o Changes to the locations and sizes of any area of localized corrosion or SCC o Changes to the size and number of any rust-colored stains resulting from iron contamination of the surface
6. Acceptance Criteria	6.1. Acceptance Criteria for General Visual Inspection of TSC Non-Welded and Non-HAZ Accessible External Surfaces: a. No evidence of cracking of any size b. No evidence of general corrosion or pitting corrosion resulting in obvious, measurable loss of base metal c. No corrosion products having a linear or branching appearance 6.2. Acceptance Criteria for TSC Welds and HAZ Areas Using VT-3: a. If no visual indications of corrosion or SCC are present (i.e. visually clean) no additional action is required. b. An assessment examination meeting the requirements of VT-1 is required if the screening examination identifies any visual anomaly that is not consistent with prior results or is identified for the first time. c. If a corrosion indication meets any of the following, it should be considered a major indication and subject to supplemental examinations per 6.4: • Cracking of any size • Corrosion products having a linear or branching appearance • Evidence of pitting corrosion, under deposit corrosion, or etching with measurable depth (removal/attack of material by corrosion)

Table A-1
AMP-1 - Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSC) (continued)

AMP Element	AMP Description
6. Acceptance Criteria (continued)	6.3. A minor indication of corrosion meets any of the following but does not meet any of the criteria for a major indication per 6.1 and 6.2.c above: • Evidence of water intrusion stained the color of corrosion products • Areas of light corrosion that follow a fabrication feature or anomaly (e.g. scratch or gouge), such indications are indicative of iron contamination • In a 10 cm × 10 cm region, corrosion product is present in less than 25% of the canister surface • Corrosion product greater than 2 mm in diameter Minor indications of corrosion within 50 mm (2inch) of a weld can be accepted by performing supplemental examinations per 6.4 to confirm that there is no CISCC present. Other minor indications are acceptable without supplemental examinations. 6.4 A supplemental examination of major indications shall be performed in accordance with Section-2400 of ASME Code Case N-860 as detailed below: a. If a surface technique is used to size a flaw, the examination shall be performed in accordance with IWA-2220 or equivalent b. If a volumetric technique is used to size a flaw, the examination shall be performed in accordance with IWA-2230 or equivalent c. If a supplemental examination is not possible or unable to provide sufficient data, an analysis shall be used when justified in accordance with N-860 Section -2440 d. The required actions of N-860 Section -2432 shall be followed, depending on the results of the supplemental examinations or N-860 Section 2441 if analysis is employed.”
7. Corrective Actions	Inspection results that do not meet the acceptance criteria are addressed under the licensee’s approved QA program. The QA program will ensure that corrective actions are completed within the licensee’s Corrective Action Program (CAP).
8. Confirmation Process	The confirmation and evaluation processes will be commensurate with the licensee’s approved QA program. The QA program will ensure that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process will describe and/or references procedures to: • Determine follow-up actions to verify effective implementation of corrective actions • Monitor for adverse trends due to recurring or repetitive findings or observations

Table A-1

AMP-1 - Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSC) (continued)

AMP Element	AMP Description
9. Administrative Controls	The administrative controls will be in accordance with the licensee's approved QA program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to NRC per 10 CFR 72.75 • frequency for updating an AMP based on site-specific, design-specific, and industrywide operational experience
10. Operating Experience	During the period of extended operation, each licensee will perform tollgate assessments of aggregated Operating Experience (OE) and other information related to the aging effects and mechanisms addressed by this AMP to determine if changes to the AMP are required to address the current state-of-knowledge. Inspection OE for NAC TSC Systems Two examinations of NAC TSCs have occurred to date: • In 2016, a TSC containing GTCC waste was inspected at Maine Yankee. The TSC did not have any reportable corrosion. It did contain a small grouping of embedded iron of no appreciable depth or height. The inspection findings included a 3 or 4 rust colored areas on the south side of the GTCC canister approximately 12 inches down from the left side of the vent. These inspection findings were evaluated in MY Condition Report CR No. 16-129, dated 7/14/16. For the 3 or 4 rust colored areas on the canister surface, each spot was approximately 1/8 inch in diameter and exhibited no depth. The areas are believed to be the result of iron contamination during original manufacturing or handling of the canister. The areas were determined to not be a concern for continued service of the canister or of affecting the canister's safety functions. • In 2018, a TSC selected to meet high susceptibility criteria containing spent fuel was inspected in accordance with the requirements of this AMP at Maine Yankee. It was considered bounding for the NAC fleet of TSCs in service. The inspection of the selected TSC did not have any reportable corrosion or SCC as documented in NAC Inspection Report No. 30013-R-01.

Table A-2
AMP-2 - Aging Management Program for Internal Vertical Concrete Casks (VCC) - Metallic Components Monitoring

AMP Element	AMP Description
1. Scope of Program	Inspection of the accessible (1) internal surfaces of steel components that are sheltered within the Vertical Concrete Casks (VCC) and managing the effects of aging. (1) The accessible surfaces of the VCC metallic internals are defined as those surfaces that can be examined using a given examination method without moving the TSC.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions.
3. Parameters Monitored/ Inspected	Parameters to be inspected and/or monitored for VCC coated steel surfaces shall include: • Visual inspection for localized corrosion resulting in significant loss of base metal. • VCC lid seal gasket (in cases where VCC lid is removed and if a gasket is installed). • Lid bolts and lid flange bolt holes (in cases where VCC lid is removed and if a gasket is installed).
4. Detection of Aging Effects	Method or Technique Aging effects are detected and characterized by: • General visual examination using direct or remote methods of the accessible VCC internal metallic components for corrosion resulting in significant loss of metal, component displacement or degradation, or air passage blockage. • The extent of inspection coverage shall be maximized, subject to the limits of accessibility. • Visual examinations shall comply with IWE-2311 requirements or their equivalent. • Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent. Sample Size These are opportunist inspections conducted in conjunction with TSC inspections. This inspection is performed when the TSC inspection is conducted. Frequency These are opportunist inspections conducted in conjunction with TSC inspections. This inspection is performed when the TSC inspection is conducted. Data Collection Documentation of the inspections required by this AMP, shall be added to the site records system in a retrievable manner. Timing These are opportunist inspections conducted in conjunction with TSC inspections. This inspection is performed when the TSC inspection is conducted.

Table A-2
AMP-2 - Aging Management Program for Internal Vertical Concrete Casks (VCC) - Metallic Components Monitoring (continued)

AMP Element	AMP Description
5. Monitoring and Trending	Monitoring and trending methods will be used to: • Establish a baseline at the beginning of the period of extended operation. • Track and trend on subsequent inspections of the selected VCC: o The appearance of the internal metallic components of the VCC will be documented to allow comparison o Changes to the locations and size of any metallic components with reportable aging effects
6. Acceptance Criteria	The acceptance criteria for the visual inspections are: • No obvious loss of base metal. • No indication of displaced or degraded components. • No indications of damaged bolts or bolt holes (in cases where VCC lid is removed). • The inspected condition of the examined area is acceptable per IWE-3511 standard or their equivalent.
7. Corrective Actions	Results that do not meet the acceptance criteria are addressed under the licensee's approved QA program. The QA program ensures that corrective actions are completed within the licensee's Corrective Action Program (CAP).
8. Confirmation Process	The confirmation process is commensurate with the licensee's QA program. The QA program ensures that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process will describe and/or references procedures to: • Determine follow-up actions to verify effective implementation of corrective actions. • Monitor for adverse trends due to recurring or repetitive findings or observations.
9. Administrative Controls	The administrative controls will be in accordance with the licensee's approved QA program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to NRC per 10 CFR 72.75 • frequency for updating an AMP based on site-specific, design-specific, and industrywide operational experience

Table A-2
AMP-2 - Aging Management Program for Internal Vertical Concrete Casks (VCC) - Metallic Components Monitoring (continued)

AMP Element	AMP Description
10. Operating Experience	During the period of extended operation, each licensee will perform tollgate assessments of aggregated Operating Experience (OE) and other information related to the aging effects and mechanisms addressed by this AMP to determine if changes to the AMP are required to address the current state-of-knowledge. <u>Inspection OE for Internal Metallic Components in NAC VCC Systems</u> Two inspections of NAC VCC systems have occurred to date. • In 2016, the internal metallic components of a NAC-UMS VCC containing a GTCC waste canister was inspected at Maine Yankee as documented in Maine Yankee Technical Evaluation MY-TE-16-005. One finding was of localized areas of coating damage on the internal VCC metallic surfaces. The finding for the VCC was localized areas of coating damage on the VCC internal areas. These are typically peeling or blistered coating areas between 1 to 4 square inches and are mostly at the corners or surface edges. The base metal appears to have minimal surface corrosion. These inspection findings were evaluated in MY Condition Report CR No. 16-129, dated 7/14/16. These conditions were determined to not be of concern in the safety functions of the VCC. • In 2018, the internal metallic components of a NAC-UMS VCC containing a SNF TSC was inspected at Maine Yankee in July 2018 as documented in NAC International Inspection Report No. 30013-R-01, Revision 0. The VCC accessible internal surfaces were inspected for localized corrosion and pitting. It was estimated that 95% of VCC accessible surfaces were inspected. During the interior VCC No 55, liner surface inspection, coating deterioration and localized corrosion (approximately 12 to 14 inches horizontally x 24 to 30 inches vertically) were identified on the liner vertical surface. The indications were evaluated by MY in Condition Report (CR) No. MY-CR-2018-128 (attached to NAC Inspection Report No. 30013-R-01). As noted in the CR, NAC performed TLAA calculation no. 30013-2002 to evaluate the conclusion that coating damage and subsequent surface corrosion as acceptable over the 60-year period of extended operation.

Table A-3
AMP-3 - Aging Management Program for External Vertical Concrete Casks (VCC) - Metallic Components Monitoring

AMP Element	AMP Description
1. Scope of Program	Inspection of the accessible external surfaces of Vertical Concrete Casks (VCC) steel components that are exposed to outdoor air and managing the effects of aging.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions.
3. Parameters Monitored/ Inspected	Parameters to be inspected and/or monitored on external VCC coated steel surfaces will include: • Visual evidence of significant coating loss or galvanic corrosion which left uncorrected could result in obvious loss of base metal. • Visual evidence of loose or missing bolts, galvanic corrosion, physical displacement, and other conditions indicative of loss of preload on VCC lid and lifting lug bolting, as applicable.
4. Detection of Aging Effects	<u>Method or Technique</u> Aging effects are detected and characterized by: • General visual examination using direct methods of the external VCC metallic components for significant corrosion or significant coating loss resulting in loss of base metal. • The extent of inspection shall cover all normally accessible VCC lid surfaces, VCC lid flange, exposed steel surfaces of the inlet and outlet vents, and VCC lid bolting. • Visual examinations shall comply with IWE-2311 requirements or their equivalent. • Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent. <u>Sample Size</u> All normally accessible and visible exterior metallic surfaces of all VCCs will be inspected. The licensee may justify alternate sample sizes based on previous inspection results. <u>Frequency</u> Inspections of readily accessible surfaces are conducted at least once every 5 years. <u>Data Collection</u> Documentation of the inspections required by this AMP, shall be added to the site records system in a retrievable manner. <u>Timing</u> The baseline inspection shall be performed within 1-year after the 20th anniversary of the first cask loaded at the ISFSI, or within 1-year after the effective date of the CoC renewal if CoC is in period of timely renewal, whichever is later.

Table A-3
AMP-3 - Aging Management Program for External Vertical Concrete Casks (VCC) - Metallic Components Monitoring (continued)

AMP Element	AMP Description
5. Monitoring and Trending	Monitoring and trending methods will be used to: • Establish a baseline at the beginning of the period of extended operation. • Track and trend on subsequent inspections of the VCC: o Changes to the locations and size of any metallic components with reportable aging effects o Location and size of areas of coating loss that could result in corrosion and obvious loss of base metal o Anomalies on the VCC lid hardware and loose bolts on VCC lid as applicable.
6. Acceptance Criteria	The acceptance criteria for the visual inspections are: • No active corrosion resulting in obvious, loss of base metal. • Areas of coating failures must remain bounded by the corrosion analysis of TLAA 30013-2002, latest revision, or are entered into the corrective action program. • No indications of loose bolts or hardware, displaced parts. • The inspected condition of the examined area is acceptable per the IWE-3511 standard or their equivalent.
7. Corrective Actions	Inspection results that do not meet the acceptance criteria are addressed under the licensee's approved QA program. The QA program ensures that corrective actions are completed within the licensee's Corrective Action Program (CAP).
8. Confirmation Process	The confirmation and evaluation processes will be commensurate with the licensee's approved QA program. The QA program will ensure that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process will describe and/or references procedures to: • Determine follow-up actions to verify effective implementation of corrective actions. • Monitor for adverse trends due to recurring or repetitive findings or observations.

Table A-3
AMP-3 - Aging Management Program for External Vertical Concrete Casks (VCC) - Metallic Components Monitoring (continued)

AMP Element	AMP Description
9. Administrative Controls	The administrative controls will be in accordance with the licensee's approved QA program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to NRC per 10 CFR 72.75 • frequency for updating an AMP based on site-specific, design-specific, and industrywide operational experience
10. Operating Experience	During the period of extended operation, each licensee will perform tollgate assessments of aggregated Operating Experience (OE) and other information related to the aging effects and mechanisms addressed by this AMP to determine if changes to the AMP are required to address the current state-of-knowledge. Inspection OE for External Metallic Components in NAC-UMS and NAC-MPC VCC Systems Thousands of these types of inspections have occurred to date on NAC-UMS and NAC-MPC VCC systems as part of the past required annual inspection provision of the applicable FSAR licensing bases. In summary: • No obvious metal loss has occurred to date on any VCC system. • Coating damage has been observed in many instances and is usually repaired in the field as part of a coating touch-up campaign. The licensee schedules this at convenient intervals and during optimum weather conditions. At no time has coating damage lead to obvious metal loss. • The external metallic components of NAC-UMS VCC No. 55 were inspected at Maine Yankee as part of pre-application inspection in accordance with the requirements of this AMP. The inspection of the selected VCC did not identify any significant corrosion or loss of base metal as documented in NAC Inspection Report No. 30013-R-01.

Table A-4
AMP-4 - Aging Management Program for Reinforced Vertical Concrete Cask (VCC) Structures – Concrete Monitoring

AMP Element	AMP Description
1. Scope of Program	General visual inspection by direct observation of the above-grade Vertical Concrete Cask (VCC) concrete structure that are directly exposed to outdoor air and managing the effects of aging.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions.
3. Parameters Monitored or Inspected	Parameters to be inspected and/or monitored for significant VCC concrete structure aging effects exceeding the acceptance criteria per ACI 349.3R-02 include the following: • Tier 3 cracking per ACI 349.3R-02. • Loss of material (spalling, scaling). • Significant porosity/permeability of concrete surfaces.
4. Detection of Aging Effects	<u>Method or Technique</u> Aging effects are detected and characterized by: • General visual inspections of the external VCC concrete surfaces using methods per ACI 349.3R-02 for cracking, loss of material, or compromised concrete integrity. • The extent of inspection coverage will include all normally accessible and visible VCC concrete surfaces. <u>Sample Size</u> All normally accessible and visible exterior concrete surfaces of all NAC VCCs in operation at the ISFSI. The licensee may justify alternate sample. <u>Frequency</u> The visual inspections of NAC VCC concrete structures will be conducted at least once every 5 years in accordance with ACI 349.3R-02 <u>Data collection</u> Documentation of the inspections required by this AMP, shall be added to the site records system in a retrievable manner. <u>Timing</u> The baseline inspection shall be performed within 1-year after the 20th anniversary of the first cask loaded at the ISFSI, or within 1-year after the effective date of the CoC renewal if CoC is in period of timely renewal, whichever is later.

Table A-4
AMP-4 - Aging Management Program for Reinforced Vertical Concrete Cask (VCC) Structures - Concrete Monitoring (continued)

AMP Element	AMP Description
5. Monitoring and Trending	Monitoring and trending methods will be used to: • Establish a baseline before or at the beginning of the period of extended operation using the 3 tier criteria of ACI 349.3R-02. • Track and trend location and size of any areas of cracking, loss of concrete material, rebar corrosion, and compromised concrete that could result in the impaired functionality and safety of the VCC.
6. Acceptance Criteria	The acceptance criteria for visual inspections are commensurate with the 3-tier criteria in ACI 349.3R-02. The following approach is utilized for inspection findings: • All tier 1 findings may be accepted without further review. • All tier 2 findings may be accepted after review by the Engineer-In-Charge. • All tier 3 findings must be reviewed by the Engineer-In-Change and are subject to further evaluations as appropriate for the finding. The type of findings addressed by the Tier 3 criteria are: • Appearance of leaching • Drummy areas that can exceed the cover concrete thickness in depth • Pop outs and voids • Scaling • Spalling • Cracks (active and passive)
7. Corrective Actions	Inspection results that do not meet the acceptance criteria are addressed under the licensee’s approved QA program. The QA program ensures that corrective actions are completed within the licensee’s Corrective Action Program (CAP).
8. Confirmation Process	The confirmation process is commensurate with the licensee’s approved QA program. The QA program ensures that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process will describe and/or reference procedures to: • Determine follow-up actions to verify effective implementation of corrective actions • Monitor for adverse trends due to recurring or repetitive findings or observations.

Table A-4
AMP-4 - Aging Management Program for Reinforced Vertical Concrete Cask (VCC) Structures - Concrete Monitoring (continued)

AMP Element	AMP Description
9. Administrative Controls	The administrative controls will be in accordance with the licensee's approved QA program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to NRC per 10 CFR 72.75 • frequency for updating an AMP based on site-specific, design-specific, and industrywide operational experience
10. Operating Experience	During the period of extended operation, each licensee will perform tollgate assessments of aggregated Operating Experience (OE) and other information related to the aging effects and mechanisms addressed by this AMP to determine if changes to the AMP are required to address the current state-of-knowledge. <u>Inspection OE for NAC-UMS and NAC-MPC VCC Concrete Structures</u> Thousands of these types of inspections have occurred to date on NAC-UMS and NAC-MPC VCC structures as part of the required annual inspection provision of the applicable FSAR licensing bases. In summary: • Tier 1, 2 and 3 passive cracking has been observed. It has been attributed to shrinkage cracking during construction. The cracks that have been trended and have not changed in size, shape or extent. • Spalling has been observed at cold weather sites. It has been attributed to the forces associated with thermal expansion differences between the concrete and the base plate and/or the prying action of freeze thaw damage. It is an active mechanism for spalling. • Efflorescence has been observed to varying degrees at different sites. It is generally considered benign and has not been associated with concrete degradation. • No staining or spalling due to rebar corrosion has been identified in the fleet.

Table A-5
AMP-5 - Aging Management Program for Transfer Casks (TFR) and Transfer Adapters

AMP Element	AMP Description
1. Scope of Program	This program manages inspections for aging effects on the accessible internal and external surfaces of steel NAC Transfer Casks (TFRs) and Transfer Adapter subcomponents that are exposed to indoor and outdoor air environments. **Note**: This AMP is not applicable to facilities not maintaining a TFR/Transfer Adapter on site. However, prior to use of a refurbished Transfer Cask and Transfer Adapter for future campaigns, the equipment shall be inspected in accordance with this AMP.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions.
3. Parameters Monitored/ Inspected	Parameters monitored or inspected for accessible TFR and Transfer Adapter surfaces include: • Visual evidence of corrosion resulting in obvious loss of base metal • Visual evidence of coating loss which left uncorrected could result in loss of base metal • Visual evidence of wear resulting in loss of base metal • Cracking or excessive wear/galling of trunnion surfaces.
4. Detection of Aging Effects	Method or Technique Aging effects are detected and characterized by: • General visual examinations using direct methods of the TFR/Transfer Adapter steel surfaces for cracking, corrosion or wear resulting in loss of base metal or coating damage which left uncorrected could result in loss of base metal. • The extent of inspection coverage will include all normally accessible and visible TFR/Transfer Adapter interior cavity and exterior surfaces. Also inspected are the retaining ring and associated bolting, shield doors and shield door rails. • Dye penetrant (PT) examinations of accessible trunnion surfaces for the presence of fatigue cracks in accordance with ASME Code, Section III, Subsection NF, NF-5350. • Visual examinations shall comply with IWE-2311 requirements or their equivalent. • Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent. Sample Size All NAC Transfer Casks/Transfer Adapters. Frequency Inspections are conducted at least once every 5 years. If a NAC TFR/Transfer Adapter is used less frequently than once every 5 years, inspections will be conducted within 1 year prior to returning the TFR/Transfer Adapter to service.

Table A-5
AMP-5 - Aging Management Program for Transfer Casks (TFRs) and Transfer Adapters (continued)

AMP Element	AMP Description
4. Detection of Aging Effects (continued)	Data Collection Documentation of the inspections required by this AMP, shall be added the site's record system in a retrievable manner. Timing Baseline inspections are completed prior to the use of the NAC TFR/Transfer Adapter in the first loading or TSC transfer campaign in the period of extended operation.
5. Monitoring and Trending	Monitoring and trending methods will be used to: • Establish a baseline during first inspection following entry into the period of extended operation • Track and trend: o Locations, size, and depth of any areas of corrosion or coating loss that could result in measurable loss of base metal o Locations of wear that results in obvious, measurable loss of base metal o Indications on TFR trunnions
6. Acceptance Criteria	For accessible surfaces, including trunnions, acceptance criteria are: • No obvious, loss of material from the base metal. • No large areas of coating failures which could expose base metal to active corrosion • No areas of wear resulting in obvious loss of base metal. • Successful completion of dye penetrant (PT) examinations of accessible trunnion surfaces for the presence of fatigue cracks in accordance with ASME Code, Section III, Subsection NF, NF-5350. • The inspected condition of the examined area is acceptable per the acceptance standards of IWE-3511 or their equivalent.
7. Corrective Actions	Results that do not meet the acceptance criteria are addressed under the licensee's approved QA program. The QA program ensures that corrective actions are completed within the licensee's Corrective Action Program (CAP).
8. Confirmation Process	The confirmation process is commensurate with the licensee's approved QA program. The QA program ensures that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process will describe or reference procedures to: • Determine follow-up actions to verify effective implementation of corrective actions. • Monitor for adverse trends due to recurring or repetitive findings or observations.

Table A-5
AMP-5 - Aging Management Program for Transfer Casks (TFRs) and Transfer Adapters (continued)

AMP Element	AMP Description
9. Administrative Controls	The administrative controls will be in accordance with the licensee's approved QA program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to NRC per 10 CFR 72.75 • frequency for updating an AMP based on site-specific, design-specific, and industrywide operational experience
10. Operating Experience	During the period of extended operation, each licensee maintaining a TFR/Transfer Adapter will perform tollgate assessments of aggregated Operating Experience (OE) and other information related to the aging effects and mechanisms addressed by this AMP to determine if changes to the AMP are required to address the current state-of-knowledge. <u>Inspection OE for NAC Transfer Casks and Transfer Adapters</u> During the periods of use of the TFRs and Transfer Adapters at the licensee's facilities, the TFRs were maintained and inspected in accordance with the requirements of ANSI N14.6. During operation of the TFRs and Transfer Adapters, areas of coating degradation were repaired by re-application of coatings. No issues with general, pitting, crevice, or galvanic corrosion have been identified. No excessive wear or loss of material has been identified on shield door to door rail to transfer adapter surfaces. No cracking of TFR lifting trunnions has been identified.

Appendix C

Final Safety Analysis Report Changed Pages and LOEP for, NAC-MPC FSAR, 21A

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C1.0 INTRODUCTION

This appendix provides a supplement and identifies pertinent changes to the NAC-MPC Updated Final Safety Analysis Report (UFSAR). Section C2.0 of this appendix contains proposed changes to the existing UFSAR. Section C3.0 of this appendix contains a proposed new Chapter 14 to the UFSAR entitled “Aging Management Program”. The new Chapter 14, Aging Management Programs, provides a summarized description of the activities for managing the effects of aging of NAC-MPC ITS systems, structures, and components. This proposed new UFSAR Chapter will also present the results of the evaluations of time-limited aging analyses (TLAAs) for the renewed license period. Chapter 14 is newly added as a result of the CoC Renewal and does not contain revision bars throughout the chapter. The headers do however indicate Revision 19A and the submittal month and year.

C2.0 CHANGES TO EXISTING UFSAR INFORMATION

List of Changes for the NAC-MPC FSAR, Revision 21A

Chapter/Page/ Figure/Table		Description of Change
Note: The List of Effective Pages and the Chapter Table of Contents, List of Figures and List of Tables have been revised accordingly to reflect the list of changes detailed below. Editorial changes made throughout the document have not been tracked.		
Chapter 1 – no changes		
Chapter 2 – no changes		
Chapter 3 – no changes		
Chapter 4 – no changes		
Chapter 5 – no changes		
Chapter 6 – no changes		
Chapter 7 – no changes		
Chapter 8 – no changes		
Chapter 9		
Page 9.2-1	Revised paragraph at the end of section 9.2	
Page 9.A.3-1	Revised paragraph at the end of section 9.A.3.1	
Page 9.A.3-2 and 9.A.3-3	Revised paragraph at the end of section 9.A.3.2	
Chapter 10 – no changes		
Chapter 11 – no changes		
Chapter 12 – no changes		
Chapter 13 – no changes		
Chapter 14		
Page 14-i thru 14.5-2	Added new chapter to address Aging Management. Revised where indicated	

C2.1 FSAR Changed Pages

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9.2 Maintenance Program

The NAC-MPC storage system is a passive system. There are no active components or systems incorporated in the design. Consequently, there is a minimal amount of maintenance that is required over its lifetime.

The system has no valves, gaskets, rupture discs or seals, and there are no accessible penetrations. Consequently, there is no maintenance associated with these types of features.

The routine thermal performance surveillance requirements for a loaded NAC-MPC system are described in the Technical Specifications of Appendix A, LCO 3.1.6 of the Certificate of Compliance.

The continuing operability of the concrete cask is verified on a 24-hour frequency by completion of SR 3.1.6.1, which allows verification by visual inspection of the inlet and outlet vents for blockage, or verification by measurement of the air temperature difference between ambient and outlet average. If the operable status of the concrete cask is reduced, the concrete cask will be returned to an operable status as specified in LCO 3.1.6.

An annual inspection of the vertical concrete cask exterior is required, and includes:

- Visual inspection of concrete surfaces for chipping, spalling or other surface defects. Any defects larger than one inch in diameter (or width) and deeper than one inch shall be regrouted, according to the grout manufacturer’s recommendations.
- Reapplication of corrosion-inhibiting (external) coatings on accessible surfaces.

An AMP for a renewed CoC commences at the end of the initial storage period for each loaded NAC-MPC System. Once the AMP has been implemented for the renewed CoC on a cask system, the performance of the AMP will replace specified maintenance inspections as detailed in Section 9.2.

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9.A.3 Maintenance Program

This section presents the maintenance requirements for the MPC-LACBWR system and the transfer cask.

9.A.3.1 MPC-LACBWR System Maintenance

The MPC-LACBWR system is a passive system. No active components or systems are incorporated in the design. Consequently, only a minimal amount of maintenance is required over its lifetime.

The MPC-LACBWR system has no valves, gaskets, rupture discs, seals, or accessible penetrations. Consequently, there is no maintenance associated with these types of features.

Annually, or on a frequency established by the user based on the environmental conditions at the ISFSI (i.e., higher inspection frequency may be appropriate at ISFSIs exposed to marine environments, lower frequency for sites located in dry environments, etc.), a program of visual inspections and maintenance of the loaded MPC-LACBWR systems in service shall be implemented. The concrete cask(s) shall be inspected as described herein.

- Visually inspect exterior concrete surfaces for chipping, spalling or other defects. Minor surface defects (i.e., approximately one cubic inch) shall be repaired by cleaning and regrouting.
- Visually inspect accessible exterior coated carbon steel surfaces for loss of coating, corrosion or other damage. The repair of corroded surfaces or surfaces missing coating materials shall be done by cleaning the areas and reapplying corrosion-inhibiting coatings in accordance with the coating manufacturer's recommendations. Exterior surface coatings authorized for use on the exposed carbon steel surfaces of concrete cask are not limited to those defined in Chapter 3 of the MPC FSAR or specified on the original design drawings. The user shall select coating appropriate to the ability to clean and recoat the affected surface areas.
- Visually inspect the installed lid bolts for presence of external corrosion. Excessively corroded, or missing, bolting shall be replaced with approved spare parts.
- Visually inspect the inlet and outlet vents to verify they are unobstructed. Remove obstructions, as necessary, to clear the vents.
- Significant damage or defects identified during the visual inspections that exceed routine maintenance shall be processed as nonconforming items

The schedule, results and corrective actions taken during the performance of the MPC-LACBWR system inspection and maintenance program shall be documented and retained as part of the system maintenance program.

An AMP for a renewed CoC commences at the end of the initial storage period for each loaded NAC-MPC System. Once the AMP has been implemented for the renewed CoC on a cask system, the performance of the AMP will replace specified maintenance inspections as detailed in Section 9.A.3.1.

9.A.3.2 Transfer Cask Maintenance

The transfer cask trunnions and shield door assemblies shall be visually inspected for gross damage and proper function prior to each use.

Annually (or a period not exceeding 14 months), an inspection and testing program shall be performed on the transfer cask in accordance with the requirements of ANSI N14.6. The following actions or alternatives shall be performed:

- Visually inspect the lifting trunnions, shield doors and shield door rails for permanent deformation and cracking. Carbon steel-coated surfaces will be inspected for chipped, cracked or missing areas of coating, and repaired by reapplication of the approved coating(s) in accordance with the coating manufacturer's recommendations.
- In addition, one of the following testing/inspection methods shall be completed.
- Perform a load test equal to or greater than 300% of the maximum service load and a post-test visual inspection of major load-bearing welds and critical components for defects, weld cracking, material displacement or permanent deformation; or
- If surface cleanliness and conditions permit, perform a dimensional and visual inspection of load-bearing components, and a nondestructive examination of major load-bearing welds.

The annual examination and testing program may be deferred during periods of nonuse of the transfer cask, provided that the transfer cask examination or testing program is performed prior to the next use of the transfer cask. The inspection results and corrective actions taken as part of the maintenance program shall be documented and retained as part of the system maintenance program.

An AMP for a renewed CoC commences at the end of the initial storage period for each loaded NAC-MPC System. Once the AMP has been implemented for the renewed CoC on a cask system, the performance of the AMP will replace specified maintenance inspections as detailed in Section 9.A.3.2.

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C3.0 NEW UFSAR CHAPTER

The following text will be integrated into the UFSAR Chapter 14 to document aging management programs credited in the license renewal review, and TLAAs evaluated to demonstrate acceptability during the period of extended operation.

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14.0 AGING MANAGEMENT

14.1 Aging Management Review

The Aging Management Review (AMR) of the NAC-MPC Storage System contained in the application for initial Certificate of Compliance (CoC) renewal provides an assessment of aging effects that could adversely affect the ability of the in-scope Structures, Systems and Components (SSCs) to perform their intended functions for the period of extended operation. The aging effects, and the mechanisms that cause them, are evaluated for the materials and storage environments. Those subcomponent of the in-scope SSCs have undergone a comprehensive review of known literature, industry operating experience (OE), NAC-MPC user OE, maintenance, and inspection records.

Aging effects that could adversely affect the ability of the in-scope SSCs to perform their safety function(s) require Aging Management Activities (AMAs) to address potential degradation during the period of extended operation. Tables 14.3-1 through Table 14.3-3 summarize those aging effects that require AMA, either by Time-Limited Aging Analyses (TLAAs) or Aging Management Programs (AMPs). The TLAAs and AMPs that are credited with managing aging effects during the period of extended operation are discussed in Sections 14.2 and 14.3, respectively.

14.2 Time-Limited Aging Analysis

A comprehensive review was conducted of the NAC-MPC design basis documents (e.g., design drawings, specifications, calculations, 72.48s, Nonconformance Reports (NCRs), and FSARs) in accordance with NUREG-1927, Revision 1 [Ref. 14.6.1] to identify and document any existing TLAAs in the original design.

For a design basis document to be considered a TLAA, all six of the following criteria taken from Reference 14.6.1 are required to be met, i.e., answered in the affirmative:

1. *Involves Structures, Systems, and Components (SSCs) important to safety within the scope of the CoC renewal.*
2. *Considers the effects of aging.*
3. *Involves time-limited assumptions defined by the current operating term of twenty (20) years.*
4. *Was determined to be relevant by NAC in making a safety determination.*
5. *Involves conclusions or provides the basis for conclusions related to the capability of the SSC to perform its intended function.*
6. *Is contained or incorporated by reference in the design basis.*

None of the NAC-MPC System design basis documents reviewed met all six criteria above. Therefore, it was concluded that there had been no TLAAs generated in the original NAC-MPC design.

As part of the CoC application for renewal, TLAAs have been prepared and incorporated into the NAC-MPC design bases for those in-scope SSCs. The additional TLAAs include: (1) Fatigue Evaluation of NAC-MPC System Components for Extended Storage; (2) Corrosion Analysis of NAC-MPC Steel Components for Extended Storage; and (3) Aging Analysis for NAC-MPC System Neutron Absorber and Neutron Shield Components (Storage/Transfer). Each of the prepared TLAAs demonstrates that the aging effect evaluated does not result in a reduction of the ability of the SSC to perform its intended safety functions for the period of extended operation as discussed in the following sections. The complete referenced calculations discussed below are included in Appendix B to the NAC-MPC CoC Renewal Application [Ref. 14.6.8].

14.2.1 Fatigue Evaluation of NAC-MPC and UMS Storage System Components for Extended Storage [Ref. 14.6.2]

The potential fatigue of the NAC-MPC SSCs (e.g., canisters and fuel baskets for YR-MPC, CY-MPC, and MPC-LACBWR systems) were evaluated in a TLAA for service conditions over the period of extended operation. The NAC-MPC canisters satisfy all conditions stipulated in NB-3222.4(d)(1) through (6), and the fuel baskets satisfy all conditions stipulated in NG-3222.4(d)(1) through (4) for a 60-year service life. Therefore, although the NAC-MPC canisters and fuel baskets do not require fatigue analysis for cyclic service for the 60-years of extended storage conditions, a TLAA has been prepared documenting why those analyses are not required.

14.2.2 Time-Limited Aging Analysis (TLAA) for Potential Corrosion of the Steel Components in the YANKEE-MPC, CY-MPC AND LACBWR-MPC Storage System VCC Assembly for a Service Life of 60-Years [Ref. 14.6.3]

The TLAA evaluated the general corrosion of NAC-MPC Vertical Concrete Cask (VCC) sheltered carbon steel components at a constant rate of 0.003-inch per year over the entire 60-year period of extended operation resulting in a total corrosion allowance of 0.18-inch. The total corrosion allowance is evaluated for the different VCC steel components and it is determined not to have an adverse effect on the ability of the VCC assembly to perform its intended structural, thermal and shielding functions. Also, there are no credible aging mechanisms that would affect the VCC steel internals to result in significant pitting or crevice corrosion. Therefore, pitting and crevice corrosion will have no adverse effects on the ability of the VCC assembly to perform its intended safety functions.

The structural evaluation of the VCC for the bottom lift by hydraulic jacks shows that the maximum bearing stress in the concrete and the maximum stresses in the pedestal with corrosion after a 60-year service life remain within the allowable stress limits. In addition, the 0.18-inch corrosion allowance on the opposite side of the plates to which the Nelson studs are welded will not adversely impact the design function of the Nelson studs. Finite element analyses of the VCC pedestals with the maximum corrosion at the end of the 60-year service period show that the maximum stress intensities in the base and ring remain well below the allowable stress limits. The margins of safety in the base and ring for the bottom lift with hydraulic jacks, with the maximum corrosion at the end of the 60-year service life, are > 10 and 3.05, respectively.

The structural evaluation of the NAC-MPC VCC for dead load, live load, flood, tornado wind, and seismic loading did not take any structural credit for the VCC steel liner, and therefore, it is concluded that any reduction in the VCC liner thickness resulting from corrosion does not change the results of the VCC analysis for these load conditions.

The structural evaluation for thermal loading concludes that a reduction of the NAC-MPC VCC steel liner thickness due to corrosion would result in a negligible change in the thermal stresses in the concrete and rebar. For the steel liner, the thermal stress is reduced due to corrosion since the reduction of the liner thickness will result in a smaller through-wall thermal gradient. Note that this reduction of the thermal gradient is greatly overshadowed by the reduction of the thermal gradient due to decay of the canister heat loads over the 60-year extended service period.

The analysis of local damage to the NAC-MPC VCC concrete shell due to tornado missile impacts did not take any structural credit for the VCC steel liner, and therefore, it is concluded that any reduction in the VCC liner thickness resulting from corrosion does not change the results of the VCC analysis for tornado missile impact. The structural evaluation of the VCC assembly for strength required to prevent perforation by the design-basis armor piercing tornado generated missile shows that the corroded lid thickness of 1.14 inches after 60-years remains sufficient to prevent missile perforation.

The structural evaluation of the NAC-MPC VCC assembly for the 6-inch drop includes an evaluation of the concrete shell and the pedestal. The evaluation of the concrete shell did not take any structural credit for the VCC steel liner, and therefore, it is concluded that any reduction in the VCC liner thickness resulting from corrosion does not change the results of the VCC concrete shell for this load conditions. The evaluation of the pedestal concluded that the maximum deformation of the pedestal due to the drop will increase to 0.69-inch, resulting in a 14% reduction of the air inlet cross-section area, which is bounded by the half inlets blocked condition. Furthermore, it is concluded that the weldment plate (and canister) will not “bottom-out” and, therefore, the canister acceleration loads will be lower than those for calculated based on the nominal plate thicknesses.

The structural evaluation of the NAC-MPC VCC assembly for the tip-over concluded that general corrosion of the steel inner shell will reduce the overall beam-bending and ring-bending stiffness of the VCC, which will slightly reduce the acceleration loads that are imparted to the canister and basket components.

The thermal analysis concludes that corrosion of the steel plates that line the VCC air passage will improve the surface properties with respect to thermal performance, but the expansion of the rust layer into the air passage could reduce the air flow cross section by up to 10%. The net effect of the corrosion of the steel surfaces that line the air passage on the thermal performance of the system is insignificant.

The NAC-MPC VCC shielding analysis concludes that the reduction in gamma shielding resulting from loss of steel due to corrosion over the extended storage period is more than offset by the decay of the source over the same timeframe.

Additionally, it has been determined that the potential impact of pitting corrosion and crevice corrosion on the performance of VCC sheltered components and determined that the potential impacts of both types of corrosion would not have a deleterious effect on the functional or safety performance of the VCC liner, shield plug, lid or baseplate/pedestal or inlet/outlet vents. The size and thickness of the carbon steel of these components and the limited number of crevices in the construction of the VCC would limit any effects of pitting and crevice corrosion on the shielding or thermal performance of the VCC carbon steel components. In addition, as both pitting and crevice corrosion would generally appear with general corrosion of the surfaces, the identification of these separate corrosion modalities would be limited by the inability to observe the surfaces below the areas of general corrosion. In conclusion, the TLAA has evaluated the ability of the VCC carbon steel components to withstand the potential uniform loss of up to 0.18 inches in thickness and still maintain the FSAR analyzed structural, thermal and shielding performance requirements and safety functions.

In addition, steel components fully encased in concrete form a thin oxide layer (passive film) due to the alkaline environment of the concrete which reduces the rate of corrosion to 0.1 μm/year. This extent of corrosion would not affect the performance of the embedded components and factors of safety are greater than 3 compared to yield strength and greater than 5 compared to ultimate strength throughout the 60-year period of extended service.

Exterior steel surfaces which are accessible (e.g., VCC lid exterior, exposed top flange surfaces, lift anchors/lift lugs and attachment components), fully exposed to the environment, and coated with primer and paint are assumed to be protected from corrosion for the 60-year period of extended operation. Therefore, stresses and factors of safety for the exposed portion of the lift anchor/lift lugs, and other coated exterior steel components remain unchanged.

14.2.3 Aging Analysis for MPC-UMS Neutron Absorber and Neutron Shield Components (Storage/Transfer) [Ref. 14.6.4]

NAC-MPC system was evaluated for:

- Depletion of the neutron absorber Boron-10 (B-10) content in the basket:
    - Considering the extremely conservative assumption of all neutrons emitted by the design basis fuel being absorbed in the neutron absorber sheets, the service life is well over 60-years.
    - A bounding depletion fraction was estimated at $1\text{x}10^{-9}$ per year. At 60-years <1% of the B-10 in the absorber sheets will be depleted.
    - There is no impact on the criticality safety of the system from such a small depletion percentage (only 75% of the minimum B-10 content is credited in the criticality analysis).
    - In a dry storage system, the neutron flux is primarily composed of non-thermal neutrons which will not deplete the neutron absorber (B-10 has primarily a thermal neutron absorption cross section).

- Depletion of the neutron absorber B-10 in the NAC-MPC system radiation shield components:
    - Considering the fluxes produced by design basis neutron sources emitted by the design basis fuel assembly, the service life in the context of boron depletion of all neutron shield components in the VCC and transfer cask is well over 60-years.
    - At 60-years <1% of the B-10 in the neutron shield will be depleted in the most limiting neutron shield component (UMS transfer cask bottom/door transfer).

- Radiation embrittlement in the cask radiation shield components:
    - Embrittlement is not a concern for the cask neutron shield components as they are captured within shells and do not perform a structural function.
    - Total gamma and neutron fluxes will not significantly impact system performance over a 60-year design life.

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14.3 Aging Management Programs

Aging effects that could result in loss of in-scope SSCs intended function(s) are managed using AMPs during the period of extended storage. The aging effects that require management are summarized in Tables 14.3-1 through 14.3-3. NAC determined for the period of extended operation there no aging effects that require aging management activity for Low Burn-up (LBU) spent fuel assemblies. There were aging effects to be considered for systems loaded with High Burn-up (HBU) spent fuel assemblies, however, NAC-MPC systems were not loaded with and are not authorized by the CoC to load HBU spent fuel assemblies. Therefore, tables for aging management activity results, either TLAAs or AMPs for spent fuel assemblies are not included in this section. Many aging effects are adequately addressed during the period of extended operation by a TLAA as discussed in Section 14.2. AMPs are used to manage those aging effects that are not addressed by a TLAA. The AMPs that manage aging effects on each of the NAC-MPC System in-scope SSCs include the following:

1. AMP 1 – Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSCs)
2. AMP 2 – Aging Management Program for Internal Vertical Concrete Cask (VCC) - Metallic Components Monitoring
3. AMP 3 – Aging Management Program for External Vertical Concrete Cask (VCC) - Metallic Components Monitoring
4. AMP 4 – Aging Management Program for Reinforced Vertical Concrete Cask (VCC) Structures - Concrete Monitoring
5. AMP 5 – Aging Management Program for Transfer Casks (TFRs) and Transfer Adapters

The AMPs for the NAC-MPC Systems are provided in Tables 14.3-4 through 14.3-8.

Table 14.3-1 Aging Management Activity Results - NAC-MPC Transportable Storage Canister (TSC) and Fuel Basket (FB)

Subcomponent	Material [(1)]	Storage Operation Environment [(2)]	Aging Mechanism	Aging Effect	Aging Management Activities Required
Shell	Stainless Steel	SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
			Fatigue	Cracking	TLAA per Design Code
	Stainless Steel (Welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
Bottom	Stainless Steel	SH	Fatigue	Cracking	TLAA per Design Code
			Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
	Stainless Steel (Welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
Shield Lid (CY and YR MPC only)	Stainless Steel	FE	Fatigue	Cracking	TLAA per Design Code
Structural Lid (CY and YR) / Closure Lid (LACBWR)	Stainless Steel	SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
			Fatigue	Cracking	TLAA per Design Code
	Stainless Steel (Welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
Port Cover (CY and YR MPC only)	Stainless Steel	FE	Fatigue	Cracking	TLAA per Design Code
Closure Ring (MPC-LACBWR only)	Stainless Steel	SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
			Fatigue	Cracking	TLAA per Design Code
	Stainless Steel (Welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP

Table 14.3-1 Aging Management Activity Results - NAC-MPC Transportable Storage Canister (TSC) and Fuel Basket (FB) (continued)

Subcomponent	**Material** (1)	**Storage Operation Environment** (2)	**Aging Mechanism**	**Aging Effect**	**Aging Management Activities Required**
Port Cover/Inner Port Cover (MPC-LACBWR only)	Stainless Steel	FE	Fatigue	Cracking	TLAA per Design Code
Outer Port Cover (MPC-LACBWR only)	Stainless Steel	SH	Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	TSC Localized Corrosion and SCC AMP
			Fatigue	Cracking	TLAA per Design Code
	Stainless Steel (Welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
Shield Lid Support Ring (CY and YR) / Lid Support Ring (MPC-LACBWR)	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
PWR / BWR Fuel Tube, Cladding, Flange	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
Neutron Absorber	Boral	HE	Boron Depletion	Loss of Criticality Control	TLAA
Bottom Fuel Basket Weldment	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
Top Fuel Basket Weldment	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
Fuel Basket Support Disk	Steel	HE	Fatigue	Cracking	TLAA per Design Code
	Stainless Steel (17-4 PH)	HE	Fatigue	Cracking	TLAA per Design Code
Fuel Basket Tie Rods, Spacers, and Washers	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
Fuel Basket Top Nut	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
Fuel Basket Flat Washer	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code

Table 14.3-1 Aging Management Activity Results - NAC-MPC Transportable Storage Canister (TSC) and Fuel Basket (FB) (continued)

Subcomponent	**Material** (1)	**Storage Operation Environment** (2)	**Aging Mechanism**	**Aging Effect**	**Aging Management Activities Required**
DFC Lid Plate and Bottom Plate	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
DFC Bottom and Side Plates	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
DFC Lid Collar and Upper Side Plates	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
DFC Tube Body	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
DFC Lift Tee, Support Ring and Dowel Pin	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code

Notes

(1) Materials Legend: Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless Steel and Stainless Steel (welded) (including precipitation hardened stainless steel); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS3); Boral = Borated aluminum-based composites; Concrete; and Spent Nuclear Fuel.

(2) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas).

Table 14.3-2 Aging Management Activity Results - NAC-MPC Vertical Concrete Cask (VCC)

Subcomponent	Material [(1)]	Storage Operation Environment [(2)]	Aging Mechanism	Aging Effect	Aging Management Activities Required
VCC Liner Shell	Steel	SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
			Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
		E-C	General Corrosion	Loss of Material	TLAA
			Pitting and Crevice Corrosion	Loss of Material	TLAA
Top Flange and Support Ring	Steel	SH	General Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
			Galvanic Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
			Pitting and Crevice Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
Base Plate Inlet Assemblies	Steel	SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
			Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
		E-C	General Corrosion	Loss of Material	TLAA
			Pitting and Crevice Corrosion	Loss of Material	TLAA
Baffle Weldment and Pedestal Plate	Steel	SH	Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
			General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
Nelson Stud	Steel	E-C	General Corrosion	Loss of Material	TLAA
			Pitting and Crevice Corrosion	Loss of Material	TLAA

Table 14.3-2 Aging Management Activity Results - NAC-MPC Vertical Concrete Cask (VCC)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Outlet Vent Assemblies	Steel	SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
			Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
		E-C	General Corrosion	Loss of Material	TLAA
			Pitting and Crevice Corrosion	Loss of Material	TLAA
VCC Lid (YR-MPC and CY-MPC) and, VCC Lid Assembly (MPC-LACBWR only)	Steel	OD	General Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
			Pitting and Crevice Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
			Galvanic Corrosion	Loss of Material	External VCC Metallic Monitoring AMP
		SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
			Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
Shield Plug Assembly (YR-MPC and CY-MPC only)	Steel	SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
			Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
	NS-3/NS-4-FR	FE	Radiation Embrittlement	Cracking	TLAA
			Thermal Aging (NS-4-FR only)	Loss of Fracture Toughness/Loss of Ductility	TLAA
			Boron Depletion (NS-4-FR only)	Loss of Shielding Effectiveness	TLAA

Table 14.3-2 Aging Management Activity Results - NAC-MPC Vertical Concrete Cask (VCC)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Rebar	Steel	E-C	Corrosion of Reinforcing Steel	Loss of Concrete/Steel Bond	TLAA
				Loss of Material (spalling, scaling)	TLAA
				Cracking	TLAA
				Loss of Strength	TLAA
Lid Bolting Hardware	Stainless Steel	SH	Galvanic Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
Concrete Shell	Concrete	OD	Reaction with Aggregates	Cracking	Reinforced VCC Structures AMP
				Loss of Strength	Reinforced VCC Structures AMP
			Salt Scaling	Loss of Material (Spalling, Scaling)	Reinforced VCC Structures AMP
			Aggressive Chemical Attack	Cracking	Reinforced VCC Structures AMP
				Loss of Strength	Reinforced VCC Structures AMP
				Loss of Material (Spalling, Scaling)	Reinforced VCC Structures AMP
			Freeze – Thaw (Above the Freeze Line)	Cracking	Reinforced VCC Structures AMP
				Loss of Material (Spalling, Scaling)	Reinforced VCC Structures AMP
			Leaching of Calcium Hydroxide	Loss of Strength	Reinforced VCC Structures AMP
				Increase in Porosity and Permeability	Reinforced VCC Structures AMP
				Reduction of Concrete pH (Reducing Corrosion Resistance of Steel Embedments)	Reinforced VCC Structures AMP

Table 14.3-2 Aging Management Activity Results - NAC-MPC Vertical Concrete Cask (VCC)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Inlet Vent Supplemental Shield Assemblies or Shield Bars (YR and LACBWR only)	Steel	SH	General Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP
			Pitting and Crevice Corrosion	Loss of Material	TLAA and Internal VCC Metallic Monitoring AMP

Notes:

(1) Materials Legend: Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless steel (including precipitation hardened SS); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS-3); Boral = Borated aluminum-based composites; Concrete; and Spent Nuclear Fuel.

(2) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas)

Table 14.3-3 Aging Management Review Results - NAC-MPC Transfer Cask (TFR)

Subcomponent	Material1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Bottom Plate	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
			Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
Inner Shell	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
			Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
Outer Shell	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
			Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
Trunnion	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
			Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
			Wear	Loss of Material	Transfer Cask AMP
Neutron Shield	NSP (NS-4-FR)	FE	Radiation Embrittlement	Cracking	TLAA
			Thermal Aging	Loss of Fracture Toughness	TLAA
			Boron Depletion	Loss of Shielding Effectiveness	TLAA
Top Plate	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
			Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
Door Rail	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
			Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
			Wear	Loss of Material	Transfer Cask AMP

Table 14.3-3 Aging Management Review Results - NAC-MPC Transfer Cask (TFR) (continued)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Retaining Ring	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
			Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
			Galvanic Corrosion	Loss of Material	Transfer Cask AMP
Retaining Ring Bolts	Stainless Steel	OD	Galvanic Corrosion	Loss of Material	Transfer Cask AMP
Shield Door Assembly	Steel	OD	Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
			Wear	Loss of Material	Transfer Cask AMP
			General Corrosion	Loss of Material	Transfer Cask AMP
Connector	Steel	OD	Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
			Wear	Loss of Material	Transfer Cask AMP
			General Corrosion	Loss of Material	Transfer Cask AMP
Strut Bracket	Steel	OD	Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
			Wear	Loss of Material	Transfer Cask AMP
			General Corrosion	Loss of Material	Transfer Cask AMP
Transfer Adapter Assembly	Steel	OD	Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
			Wear	Loss of Material	Transfer Cask AMP
			General Corrosion	Loss of Material	Transfer Cask AMP

Notes:

(1) Materials Legend: Steel (Including various carbon, alloy, high-strength, and low-alloy steels. Also includes galvanized and electroless nickel (EN) plated steels); Stainless steel (including precipitation hardened SS); Aluminum; NSP = Polymer-Based Neutron Shielding (e.g., NS-4-FR); NSC = Cement-Based Neutron shielding (e.g., NS3); Lead; Boral = Borated aluminum-based composites (Boral); Concrete; and SNF = Spent Nuclear Fuel

(2) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas).

Table 14.3-4 AMP-1 - Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSC)

AMP Element	AMP Description
1. Program Scope	Examination of welded stainless-steel dry storage Transportable Storage Canisters (TSC) readily accessible (1) external surfaces for localized corrosion and stress corrosion cracking (SCC). (1) The accessible surfaces of the TSC are defined as those surfaces that can be examined using a given examination method without moving the TSC.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions.
3. Parameters Monitored/ Inspected	Parameters monitored and/or inspected include: • Visual evidence of localized corrosion, including pitting corrosion and crevice corrosion, and SCC. • Size and location of localized corrosion and SCC on TSC welds and heat affected zones (HAZs) (≤ 2 inches [50mm] from weld edge). • Appearance and location of discontinuities on the examined TSC surfaces.
4. Detection of Aging Effects	Method or Technique Aging effects are detected and characterized by: • General visual examination using direct or remote methods of the TSC accessible external surfaces away from the weld region for localized corrosion and anomalies. • Visual examination by direct or remote means of accessible TSC welds, associated HAZs, and known areas of removed temporary attachments and weld repairs using qualified VT-3 methods and equipment to identify corrosion products that may be indicators of localized corrosion and SCC. • An assessment examination meeting the requirements of VT-1 is required if the screening examination identifies any visual anomaly that is not consistent with prior results or is identified for the first time. • A supplemental examination is required for any visual anomaly within the weld region that is classified as a major indication as discussed in Section 6, Acceptance Criteria. • The extent of coverage shall be maximized subject to the limits of accessibility. Sample Size For sites conducting a TSC examination there should be a minimum of one TSC examined at each site. Preference should be given to the TSC(s) with the greatest susceptibility for localized corrosion or SCC. Frequency • Baseline inspection at beginning of the period of extended operation • Every 10 years for TSCs without detection of indications of major corrosion degradation or SCC • Every 5 years for TSCs with detection of major indications of corrosion degradation or detection(s) of SCC

Table 14.3-4 AMP-1 - Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSC)

AMP Element	AMP Description
4. Detection of Aging Effects (continued)	Data Collection Documentation of the examination of the TSC, location and appearance of deposits, and an assessment of the suspect areas where corrosion products and/or SCC were observed as described in corrective actions shall be maintained in the licensee's record retention system. Timing of Inspections The baseline inspection shall be performed within 1-year after the 20th anniversary of the first cask loaded at the ISFSI, or within 1-year after the effective date of the CoC renewal if CoC is in period of timely renewal, whichever is later unless otherwise justified.
5. Monitoring and Trending	Monitoring and trending methods will: • Establish a baseline at the beginning of the period of extended operation for the selected TSC. • Track and trend on subsequent inspections of the selected TSC: o The appearance of the selected TSC, particularly at welds and crevice locations documented with images and/or video that will allow comparison o Changes to the locations and sizes of any area of localized corrosion or SCC o Changes to the size and number of any rust-colored stains resulting from iron contamination of the surface
6. Acceptance Criteria	6.1. Acceptance Criteria for General Visual Inspection of TSC Non-Welded and Non-HAZ Accessible External Surfaces: a. No evidence of cracking of any size b. No evidence of general corrosion or pitting corrosion resulting in obvious, measurable loss of base metal c. No corrosion products having a linear or branching appearance 6.2. Acceptance Criteria for TSC Welds and HAZ Areas Using VT-3: a. If no visual indications of corrosion or SCC are present (i.e. visually clean) no additional action is required. b. An assessment examination meeting the requirements of VT-1 is required if the screening examination identifies any visual anomaly that is not consistent with prior results or is identified for the first time. c. If a corrosion indication meets any of the following, it should be considered a major indication and subject to supplemental examinations per 6.4: • Cracking of any size • Corrosion products having a linear or branching appearance • Evidence of pitting corrosion, under deposit corrosion, or etching with measurable depth (removal/attack of material by corrosion)

Table 14.3-4 AMP-1 - Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSC)

AMP Element	AMP Description
6. Acceptance Criteria (continued)	6.3. A minor indication of corrosion meets any of the following but does not meet any of the criteria for a major indication per 6.1 and 6.2.c above: • Evidence of water intrusion stained the color of corrosion products • Areas of light corrosion that follow a fabrication feature or anomaly (e.g. scratch or gouge), such indications are indicative of iron contamination • In a 10 cm × 10 cm region, corrosion product is present in less than 25% of the canister surface • Corrosion product greater than 2 mm in diameter Minor indications of corrosion within 50 mm (2inch) of a weld can be accepted by performing supplemental examinations per 6.4 to confirm that there is no CISCC present. Other minor indications are acceptable without supplemental examinations. 6.4 A supplemental examination of major indications shall be performed in accordance with Section-2400 of ASME Code Case N-860 as detailed below: a. If a surface technique is used to size a flaw, the examination shall be performed in accordance with IWA-2220 or equivalent b. If a volumetric technique is used to size a flaw, the examination shall be performed in accordance with IWA-2230 or equivalent c. If a supplemental examination is not possible or unable to provide sufficient data, an analysis shall be used when justified in accordance with N-860 Section -2440 d. The required actions of N-860 Section -2432 shall be followed, depending on the results of the supplemental examinations or N-860 Section 2441 if analysis is employed.”
7. Corrective Actions	Inspection results that do not meet the acceptance criteria are addressed under the licensee’s approved QA program. The QA program will ensure that corrective actions are completed within the licensee’s Corrective Action Program (CAP).
8. Confirmation Process	The confirmation and evaluation processes will be commensurate with the licensee’s approved QA program. The QA program will ensure that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process will describe and/or references procedures to: • Determine follow-up actions to verify effective implementation of corrective actions • Monitor for adverse trends due to recurring or repetitive findings or observations

Table 14.3-4 AMP-1 - Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSC)

AMP Element	AMP Description
9. Administrative Controls	The administrative controls will be in accordance with the licensee's approved QA program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to NRC per 10 CFR 72.75 • frequency for updating an AMP based on site-specific, design-specific, and industrywide operational experience
10. Operating Experience	During the period of extended operation, each licensee will perform tollgate assessments of aggregated Operating Experience (OE) and other information related to the aging effects and mechanisms addressed by this AMP to determine if changes to the AMP are required to address the current state-of-knowledge. Inspection OE for NAC TSC Systems Two examinations of NAC TSCs have occurred to date: • In 2016, a TSC containing GTCC waste was inspected at Maine Yankee. The TSC did not have any reportable corrosion. It did contain a small grouping of embedded iron of no appreciable depth or height. The inspection findings included a 3 or 4 rust colored areas on the south side of the GTCC canister approximately 12 inches down from the left side of the vent. These inspection findings were evaluated in MY Condition Report CR No. 16-129, dated 7/14/16. For the 3 or 4 rust colored areas on the canister surface, each spot was approximately 1/8 inch in diameter and exhibited no depth. The areas are believed to be the result of iron contamination during original manufacturing or handling of the canister. The areas were determined to not be a concern for continued service of the canister or of affecting the canister's safety functions. • In 2018, a TSC selected to meet high susceptibility criteria containing spent fuel was inspected in accordance with the requirements of this AMP at Maine Yankee. It was considered bounding for the NAC fleet of TSCs in service. The inspection of the selected TSC did not have any reportable corrosion or SCC as documented in NAC Inspection Report No. 30013-R-01.

Table 14.3-5 AMP-2 - Aging Management Program for Internal Vertical Concrete Casks (VCC) Metallic Components Monitoring

AMP Element	AMP Description
1. Scope of Program	Inspection of the accessible [1] internal surfaces of steel components that are sheltered within the Vertical Concrete Casks (VCC) and managing the effects of aging. (1) The accessible surfaces of the VCC metallic internals are defined as those surfaces that can be examined using a given examination method without moving the TSC.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions.
3. Parameters Monitored/ Inspected	Parameters to be inspected and/or monitored for VCC coated steel surfaces shall include: • Visual inspection for localized corrosion resulting in significant loss of base metal. • VCC lid seal gasket (in cases where VCC lid is removed and if a gasket is installed). • Lid bolts and lid flange bolt holes (in cases where VCC lid is removed and if a gasket is installed).
4. Detection of Aging Effects	Method or Technique Aging effects are detected and characterized by: • General visual examination using direct or remote methods of the accessible VCC internal metallic components for corrosion resulting in significant loss of metal, component displacement or degradation, or air passage blockage. • The extent of inspection coverage shall be maximized, subject to the limits of accessibility. • Visual examinations shall comply with IWE-2311 requirements or their equivalent. • Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent. Sample Size These are opportunist inspections conducted in conjunction with TSC inspections. This inspection is performed when the TSC inspection is conducted. Frequency These are opportunist inspections conducted in conjunction with TSC inspections. This inspection is performed when the TSC inspection is conducted. Data Collection Documentation of the inspections required by this AMP, shall be added to the site records system in a retrievable manner. Timing These are opportunist inspections conducted in conjunction with TSC inspections. This inspection is performed when the TSC inspection is conducted.

Table 14.3-5 AMP-2 - Aging Management Program for Internal Vertical Concrete Casks (VCC) Metallic Components Monitoring

AMP Element	AMP Description
5. Monitoring and Trending	Monitoring and trending methods will be used to: • Establish a baseline at the beginning of the period of extended operation. • Track and trend on subsequent inspections of the selected VCC: o The appearance of the internal metallic components of the VCC will be documented to allow comparison o Changes to the locations and size of any metallic components with reportable aging effects
6. Acceptance Criteria	The acceptance criteria for the visual inspections are: • No obvious loss of base metal. • No indication of displaced or degraded components. • No indications of damaged bolts or bolt holes (in cases where VCC lid is removed). • The inspected condition of the examined area is acceptable per IWE-3511 standard or their equivalent.
7. Corrective Actions	Results that do not meet the acceptance criteria are addressed under the licensee's approved QA program. The QA program ensures that corrective actions are completed within the licensee's Corrective Action Program (CAP).
8. Confirmation Process	The confirmation process is commensurate with the licensee's QA program. The QA program ensures that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process will describe and/or references procedures to: • Determine follow-up actions to verify effective implementation of corrective actions. • Monitor for adverse trends due to recurring or repetitive findings or observations.
9. Administrative Controls	The administrative controls will be in accordance with the licensee's approved QA program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to NRC per 10 CFR 72.75 • frequency for updating an AMP based on site-specific, design-specific, and industrywide operational experience

Table 14.3-5 AMP-2 - Aging Management Program for Internal Vertical Concrete Casks (VCC) Metallic Components Monitoring

AMP Element	AMP Description
10. Operating Experience	During the period of extended operation, each licensee will perform tollgate assessments of aggregated Operating Experience (OE) and other information related to the aging effects and mechanisms addressed by this AMP to determine if changes to the AMP are required to address the current state-of-knowledge. <u>Inspection OE for Internal Metallic Components in NAC VCC Systems</u> Two inspections of NAC VCC systems have occurred to date. • In 2016, the internal metallic components of a NAC-UMS VCC containing a GTCC waste canister was inspected at Maine Yankee as documented in Maine Yankee Technical Evaluation MY-TE-16-005. One finding was of localized areas of coating damage on the internal VCC metallic surfaces. The finding for the VCC was localized areas of coating damage on the VCC internal areas. These are typically peeling or blistered coating areas between 1 to 4 square inches and are mostly at the corners or surface edges. The base metal appears to have minimal surface corrosion. These inspection findings were evaluated in MY Condition Report CR No. 16-129, dated 7/14/16. These conditions were determined to not be of concern in the safety functions of the VCC. • In 2018, the internal metallic components of a NAC-UMS VCC containing a SNF TSC was inspected at Maine Yankee in July 2018 as documented in NAC International Inspection Report No. 30013-R-01, Revision 0. The VCC accessible internal surfaces were inspected for localized corrosion and pitting. It was estimated that 95% of VCC accessible surfaces were inspected. During the interior VCC No 55 liner surface inspection, coating deterioration and localized corrosion (approximately 12 to 14 inches horizontally x 24 to 30 inches vertically) were identified on the liner vertical surface. The indications were evaluated by MY in Condition Report (CR) No. MY-CR-2018-128 (attached to NAC Inspection Report No. 30013-R-01). As noted in the CR, NAC performed TLAA calculation no. 30013-2002 to evaluate the conclusion that coating damage and subsequent surface corrosion as acceptable over the 60 year period of extended operation.

Table 14.3-6 AMP-3 - Aging Management Program for External Vertical Concrete Casks (VCC) Metallic Components Monitoring

AMP Element	AMP Description
1. Scope of Program	Inspection of the accessible external surfaces of Vertical Concrete Casks (VCC) steel components that are exposed to outdoor air and managing the effects of aging.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions.
3. Parameters Monitored/ Inspected	Parameters to be inspected and/or monitored on external VCC coated steel surfaces will include: • Visual evidence of significant coating loss or galvanic corrosion which left uncorrected could result in obvious loss of base metal. • Visual evidence of loose or missing bolts, galvanic corrosion, physical displacement, and other conditions indicative of loss of preload on VCC lid and lifting lug bolting, as applicable.
4. Detection of Aging Effects	Method or Technique Aging effects are detected and characterized by: • General visual examination using direct methods of the external VCC metallic components for significant corrosion or significant coating loss resulting in loss of base metal. • The extent of inspection shall cover all normally accessible VCC lid surfaces, VCC lid flange, exposed steel surfaces of the inlet and outlet vents, and VCC lid bolting. • Visual examinations shall comply with IWE-2311 requirements or their equivalent. • Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent. Sample Size All normally accessible and visible exterior metallic surfaces of all VCCs will be inspected. The licensee may justify alternate sample sizes based on previous inspection results. Frequency Inspections of readily accessible surfaces are conducted at least once every 5 years. Data Collection Documentation of the inspections required by this AMP, shall be added to the site records system in a retrievable manner. Timing The baseline inspection shall be performed within 1-year after the 20th anniversary of the first cask loaded at the ISFSI, or within 1-year after the effective date of the CoC renewal if CoC is in period of timely renewal, whichever is later.

Table 14.3-6 AMP-3 - Aging Management Program for External Vertical Concrete Casks (VCC) Metallic Components Monitoring

AMP Element	AMP Description
5. Monitoring and Trending	Monitoring and trending methods will be used to: • Establish a baseline at the beginning of the period of extended operation. • Track and trend on subsequent inspections of the VCC: o Changes to the locations and size of any metallic components with reportable aging effects o Location and size of areas of coating loss that could result in corrosion and obvious loss of base metal o Anomalies on the VCC lid hardware and loose bolts on VCC lid as applicable.
6. Acceptance Criteria	The acceptance criteria for the visual inspections are: • No active corrosion resulting in obvious, loss of base metal. • Areas of coating failures must remain bounded by the corrosion analysis of TLAA 30013-2002, latest revision, or are entered into the corrective action program. • No indications of loose bolts or hardware, displaced parts. • The inspected condition of the examined area is acceptable per the IWE-3511 standard or their equivalent.
7. Corrective Actions	Inspection results that do not meet the acceptance criteria are addressed under the licensee's approved QA program. The QA program ensures that corrective actions are completed within the licensee's Corrective Action Program (CAP).
8. Confirmation Process	The confirmation and evaluation processes will be commensurate with the licensee's approved QA program. The QA program will ensure that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process will describe and/or references procedures to: • Determine follow-up actions to verify effective implementation of corrective actions. • Monitor for adverse trends due to recurring or repetitive findings or observations.

Table 14.3-6 AMP-3 - Aging Management Program for External Vertical Concrete Casks (VCC) Metallic Components Monitoring

AMP Element	AMP Description
9. Administrative Controls	The administrative controls will be in accordance with the licensee's approved QA program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to NRC per 10 CFR 72.75 • frequency for updating an AMP based on site-specific, design-specific, and industrywide operational experience
10. Operating Experience	During the period of extended operation, each licensee will perform tollgate assessments of aggregated Operating Experience (OE) and other information related to the aging effects and mechanisms addressed by this AMP to determine if changes to the AMP are required to address the current state-of-knowledge. <u>Inspection OE for External Metallic Components in NAC-UMS and NAC-MPC VCC Systems</u> Thousands of these types of inspections have occurred to date on NAC-UMS and NAC-MPC VCC systems as part of the past required annual inspection provision of the applicable FSAR licensing bases. In summary: • No obvious metal loss has occurred to date on any VCC system. • Coating damage has been observed in many instances and is usually repaired in the field as part of a coating touch-up campaign. The licensee schedules this at convenient intervals and during optimum weather conditions. At no time has coating damage lead to obvious metal loss. • The external metallic components of NAC-UMS VCC No. 55 were inspected at Maine Yankee as part of pre-application inspection in accordance with the requirements of this AMP. The inspection of the selected VCC did not identify any significant corrosion or loss of base metal as documented in NAC Inspection Report No. 30013-R-01.

Table 14.3-7 AMP-4 - Aging Management Program for NAC Reinforced Vertical Concrete Cask (VCC) Structures

AMP Element	AMP Description
1. Scope of Program	General visual inspection by direct observation of the above-grade Vertical Concrete Cask (VCC) concrete structure that are directly exposed to outdoor air and managing the effects of aging.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions.
3. Parameters Monitored or Inspected	Parameters to be inspected and/or monitored for significant VCC concrete structure aging effects exceeding the acceptance criteria per ACI 349.3R-02 include the following: • Tier 3 cracking per ACI 349.3R-02. • Loss of material (spalling, scaling). • Significant porosity/permeability of concrete surfaces.
4. Detection of Aging Effects	<u>Method or Technique</u> Aging effects are detected and characterized by: • General visual inspections of the external VCC concrete surfaces using methods per ACI 349.3R-02 for cracking, loss of material, or compromised concrete integrity. • The extent of inspection coverage will include all normally accessible and visible VCC concrete surfaces. <u>Sample Size</u> All normally accessible and visible exterior concrete surfaces of all NAC VCCs in operation at the ISFSI. The licensee may justify alternate sample sizes. <u>Frequency</u> The visual inspections of NAC VCC concrete structures will be conducted at least once every 5 years in accordance with ACI 349.3R-02 <u>Data collection</u> Documentation of the inspections required by this AMP, shall be added to the site records system in a retrievable manner. <u>Timing</u> The baseline inspection shall be performed within 1-year after the 20th anniversary of the first cask loaded at the ISFSI, or within 1-year after the effective date of the CoC renewal if CoC is in period of timely renewal, whichever is later.

Table 14.3-7 AMP-4 - Aging Management Program for NAC Reinforced Vertical Concrete Cask (VCC) Structures

AMP Element	AMP Description
5. Monitoring and Trending	Monitoring and trending methods will be used to: • Establish a baseline before or at the beginning of the period of extended operation using the 3 tier criteria of ACI 349.3R-02. • Track and trend location and size of any areas of cracking, loss of concrete material, rebar corrosion, and compromised concrete that could result in the impaired functionality and safety of the VCC.
6. Acceptance Criteria	The acceptance criteria for visual inspections are commensurate with the 3-tier criteria in ACI 349.3R-02. The following approach is utilized for inspection findings: • All tier 1 findings may be accepted without further review. • All tier 2 findings may be accepted after review by the Engineer-In-Charge. • All tier 3 findings must be reviewed by the Engineer-In-Change and are subject to further evaluations as appropriate for the finding. The type of findings addressed by the Tier 3 criteria are: • Appearance of leaching • Drummy areas that can exceed the cover concrete thickness in depth • Pop outs and voids • Scaling • Spalling • Cracks (active and passive)
7. Corrective Actions	Inspection results that do not meet the acceptance criteria are addressed under the licensee's approved QA program. The QA program ensures that corrective actions are completed within the licensee's Corrective Action Program (CAP).
8. Confirmation Process	The confirmation process is commensurate with the licensee's approved QA program. The QA program ensures that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process will describe and/or reference procedures to: • Determine follow-up actions to verify effective implementation of corrective actions. • Monitor for adverse trends due to recurring or repetitive findings or observations.

Table 14.3-7 AMP-4 - Aging Management Program for NAC Reinforced Vertical Concrete Cask (VCC) Structures

AMP Element	AMP Description
9. Administrative Controls	The administrative controls will be in accordance with the licensee's approved QA program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to NRC per 10 CFR 72.75 • frequency for updating an AMP based on site-specific, design-specific, and industrywide operational experience
10. Operating Experience	During the period of extended operation, each licensee will perform tollgate assessments of aggregated Operating Experience (OE) and other information related to the aging effects and mechanisms addressed by this AMP to determine if changes to the AMP are required to address the current state-of-knowledge. <u>Inspection OE for NAC-UMS and NAC-MPC VCC Concrete Structures</u> Thousands of these types of inspections have occurred to date on NAC-UMS and NAC-MPC VCC structures as part of the required annual inspection provision of the applicable FSAR licensing bases. In summary: • Tier 1, 2 and 3 passive cracking has been observed. It has been attributed to shrinkage cracking during construction. The cracks that have been trended and have not changed in size, shape or extent. • Spalling has been observed at cold weather sites. It has been attributed to the forces associated with thermal expansion differences between the concrete and the base plate and/or the prying action of freeze thaw damage. It is an active mechanism for spalling. • Efflorescence has been observed to varying degrees at different sites. It is generally considered benign and has not been associated with concrete degradation. • No staining or spalling due to rebar corrosion has been identified in the fleet.

Table 14.3-8 AMP-5 - Aging Management Program for Transfer Casks (TFR) and Transfer Adapters

AMP Element	AMP Description
1. Scope of Program	This program manages inspections for aging effects on the accessible internal and external surfaces of steel NAC Transfer Casks (TFRs) and Transfer Adapter subcomponents that are exposed to indoor and outdoor air environments. **Note**: This AMP is not applicable to facilities not maintaining a TFR/Transfer Adapter on site. However, prior to use of a refurbished Transfer Cask and Transfer Adapter for future campaigns, the equipment shall be inspected in accordance with this AMP.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions.
3. Parameters Monitored/ Inspected	Parameters monitored or inspected for accessible TFR and Transfer Adapter surfaces include: • Visual evidence of corrosion resulting in obvious loss of base metal • Visual evidence of coating loss which left uncorrected could result in loss of base metal • Visual evidence of wear resulting in loss of base metal • Cracking or excessive wear/galling of trunnion surfaces.
4. Detection of Aging Effects	Method or Technique Aging effects are detected and characterized by: • General visual examinations using direct methods of the TFR/Transfer Adapter steel surfaces for cracking, corrosion or wear resulting in loss of base metal or coating damage which left uncorrected could result in loss of base metal. • The extent of inspection coverage will include all normally accessible and visible TFR/Transfer Adapter interior cavity and exterior surfaces. Also inspected are the retaining ring and associated bolting, shield doors and shield door rails. • Dye penetrant (PT) examinations of accessible trunnion surfaces for the presence of fatigue cracks in accordance with ASME Code, Section III, Subsection NF, NF-5350. • Visual examinations shall comply with IWE-2311 requirements or their equivalent. • Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent. Sample Size All NAC Transfer Casks/Transfer Adapters. Frequency Inspections are conducted at least once every 5 years. If a NAC TFR/Transfer Adapter is used less frequently than once every 5 years, inspections will be conducted within 1 year prior to returning the TFR/Transfer Adapter to service.

Table 14.3-8 AMP-5 - Aging Management Program for Transfer Casks (TFR) and Transfer Adapters

Element	Description
4. Detection of Aging Effects (continued)	<u>Data Collection</u> Documentation of the inspections required by this AMP, shall be added the site's record system in a retrievable manner. <u>Timing</u> Baseline inspections are completed prior to the use of the NAC TFR/Transfer Adapter in the first loading or TSC transfer campaign in the period of extended operation.
5. Monitoring and Trending	Monitoring and trending methods will be used to: • Establish a baseline during first inspection following entry into the period of extended operation • Track and trend: o locations, size, and depth of any areas of corrosion or coating loss that could result in measurable loss of base metal o locations of wear that results in obvious, measurable loss of base metal o indications on TFR trunnions
6. Acceptance Criteria	For accessible surfaces, including trunnions, acceptance criteria are: • No obvious, loss of material from the base metal. • No large areas of coating failures which could expose base metal to active corrosion • No areas of wear resulting in obvious loss of base metal. • Successful completion of dye penetrant (PT) examinations of accessible trunnion surfaces for the presence of fatigue cracks in accordance with ASME Code, Section III, Subsection NF, NF-5350. • The inspected condition of the examined area is acceptable per the acceptance standards of IWE-3511 or their equivalent.
7. Corrective Actions	Results that do not meet the acceptance criteria are addressed under the licensee's approved QA program. The QA program ensures that corrective actions are completed within the licensee's Corrective Action Program (CAP).
8. Confirmation Process	The confirmation process is commensurate with the licensee's approved QA program. The QA program ensures that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality. The confirmation process will describe or reference procedures to: • Determine follow-up actions to verify effective implementation of corrective actions. • Monitor for adverse trends due to recurring or repetitive findings or observations.

Table 14.3-8 AMP-5 - Aging Management Program for Transfer Casks (TFR) and Transfer Adapters

Element	Description
9. Administrative Controls	The administrative controls will be in accordance with the licensee's approved QA program approved under 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B, respectively. The QA program ensures that administrative controls include provisions that define: • instrument calibration and maintenance • inspector requirements • record retention requirements • document control The administrative controls describe or reference: • methods for reporting results to NRC per 10 CFR 72.75 • frequency for updating an AMP based on site-specific, design-specific, and industrywide operational experience
10. Operating Experience	During the period of extended operation, each licensee maintaining a TFR/Transfer Adapter will perform tollgate assessments of aggregated Operating Experience (OE) and other information related to the aging effects and mechanisms addressed by this AMP to determine if changes to the AMP are required to address the current state-of-knowledge. Inspection OE for NAC Transfer Casks and Transfer Adapters During the periods of use of the TFRs and Transfer Adapters at the licensee's facilities, the TFRs were maintained and inspected in accordance with the requirements of ANSI N14.6. During operation of the TFRs and Transfer Adapters, areas of coating degradation were repaired by re-application of coatings. No issues with general, pitting, crevice, or galvanic corrosion have been identified. No excessive wear or loss of material has been identified on shield door to door rail to transfer adapter surfaces. No cracking of TFR lifting trunnions has been identified.

14.4 Retrievability

Retrievability is the ability to readily retrieve spent nuclear fuel from storage for further processing or disposal in accordance with 10 CFR 72.122 (l). ISG-2, Revision 2 [14.6.5] provides staff guidance on the subject of ready retrieval as "the ability to safely remove the spent fuel from storage for further processing or disposal." Per ISG-2, the NRC interprets this regulation that a storage system be designed to allow ready retrieval in the initial design, amendments to the design, and in license renewal, through the aging management of the design.

In order to demonstrate the ability for ready retrieval, a licensee should demonstrate it has the ability to perform any of the three options listed below. These options may be utilized individually or in any combination or sequence, as appropriate.

A. Remove individual or canned spent fuel assemblies from wet or dry storage,
B. Remove a canister loaded with spent fuel assemblies from a storage cask/overpack,
C. Remove a cask loaded with spent fuel assemblies from the storage location.

The NAC-MPC storage system is designed to allow ready retrieval of the SNF assemblies for further processing and disposal, in accordance with 10 CFR 72.122(l) by either option A. or option B above. Under Option A, the NAC-MPC canisters are designed for opening of the canister at a suitable facility for removal and transfer of the individual or canned spent fuel assemblies, and under Option B by transfer of a loaded NAC-MPC canister to the approved and NRC certified NAC-STC transport cask system (CoC No. 71-9235) [Ref. 14.6.6] for transport off-site without the need for repackaging.

The results of the AMR show there are no credible aging effects in the SNF assemblies that require management during the period of extended storage. Only low burnup (≤ 45 GWd/MTU), intact and damaged (loaded in damaged fuel cans [DFCs]), zircaloy and stainless steel clad PWR and BWR SNF assemblies are stored in the NAC-MPC storage system. Degradation of the cladding of low burnup fuel will not occur during the period of extended operation because the inert helium atmosphere inside the canister is maintained. Corrosion and chloride-induced stress corrosion cracking (CISCC) of the canister, and canister lid and confinement welds and heat affected zones (HAZs) is managed by an AMP during the period of extended operation to ensure that no aging effect will result in the loss of their intended primary safety functions of confinement and structural integrity. Therefore, ready retrieval of the SNF is maintained during the period of extended operation by maintaining the structural integrity of the NAC-MPC canister to be lifted and transferred to a NAC-STC transport cask. During the AMR, the appropriate NAC-MPC canister components required for the ready retrieval of the SNF and/or canister have been identified as components required to maintain retrievability and identified as RE in the AMR tables in the CoC Renewal Application.

These efforts provide reasonable assurance that the SFAs will be capable of being removed from the canister by normal means or that the canister can be directly transferred to a certified NAC-STC transport cask for off-site transport.

14.5 Periodic Tollgate Assessments

Periodic tollgate assessments are part of a process known as operations-based aging management which is described in NEI 14-03 (Ref. 14.6.7). To implement this process for the NAC-MPC System, general licensees will perform written evaluations periodically for each AMP in use at their ISFSI. The tollgate assessments are a method for the general licensee to evaluate the AMP, the results from AMP inspections and the aggregate impact of aging-related dry cask storage system OE, research, monitoring, and inspections on the aging management of in-scope SSCs. Tollgate assessments are intended to include non-nuclear and international operating information on a best-effort basis. Tollgate assessments may be authored by entities other than the general licensee but remain the general licensee's responsibility. Corrective actions arising from tollgate assessments are managed through the corrective action program of the licensee and/or the CoC General licensees have tollgate assessment responsibilities, as discussed below.

14.5.1 Tollgate Assessments by General Licensees

As detailed in Table 14.5-1, during the twenty-fifth calendar year following initial loading of a MPC dry storage system beginning STORAGE OPERATIONS, the general licensee shall conduct and document an initial tollgate assessment for each AMP in use, which should address the following areas:

- A summary of research findings, OE, monitoring data, and inspection results
- Aggregate impact of findings (including trends)
- Consistency with assumptions and inputs in TLAAs
- Effectiveness of AMPs
- Corrective actions, including changes to AMPs or TLAAs
- Summary and conclusions

Appendix A of NEI 14-03 (Ref. 14.6.7) provides performance criteria for assessing AMPs and should be used for further guidance. Subsequent periodic tollgate assessments are also discussed in Table 14.5-1 and this guidance should be used to determine the timing of these subsequent assessments.

Tollgate assessments will generally result in one of three conclusions:

1. The information reviewed confirms the adequacy of current TLAAs and AMPs. Continued safe storage is expected to the next tollgate.
2. Information is currently unavailable for a potential aging-related degradation mechanism. Plans to address the information gap should be developed and implemented.

3. The industry information reviewed introduces issues not adequately managed by current TLAAs and AMPs. Corrective actions are required. This could be as simple as changes to the TLAAs or AMPs, as appropriate, or could involve additional inspections, mitigation, repairs, or replacements of DSS components.

14.5.2 Specific Requirements for Aging Management Tollgates

14.5.2.1 Introduction

AMPs are defined in Tables 14.3-4 through 14.3-8 for the localized corrosion and stress corrosion cracking of welded transportable storage canisters; internal vertical concrete cask metallic components monitoring; external vertical concrete cask metallic components monitoring; reinforced vertical concrete cask concrete structures concrete monitoring; and transfer casks and transfer adapters. These AMPs are subject to modification under 10 CFR 72.48 as improvements are identified by tollgate assessments or the corrective action program.

14.5.2.2 Generic Tollgate Process

Using the guidance found in section 14.5.1 and Table 14.5-1, the AMPs defined in Tables 14.3-4 through 14.3-8 are subject to initial and periodic tollgate assessments. Assessments are not stopping points. No action other than performing an assessment is required to continue NAC-MPC STORAGE OPERATIONS. The tollgate process applies only to those licensees for whom the corresponding AMP applies. Tollgate assessment reports are not required to be submitted to the NRC but must be available for inspection as a required aging management record.

Upon completion of these assessments, corrective actions may be required to continue to ensure that in-scope SSCs perform their safety function or to improve the AMP performance criteria (Appendix A of NEI 14-03).

Corrective actions may include:

- Modification of an AMP or TLAA
- Adjustment of the scope, frequency, or both of AMPs
- Repair or replacement of SSCs

Table 14.5-1 TSC AMP for the Effects of SCC Tollgates

Tollgate	Home	Assessment
1	T_0+ 5 yrs (1)	Evaluate information from the industry sources on a best-effort basis 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 AMP with which they are associated with: • Results, if any, of research and development programs focused specifically on aging related degradation mechanism identified as potentially affecting the UMS storage system, such as those conducted by Electric Power Research Institute (EPRI), the Department of Energy (DOE), and DOE/University programs. • Results of prior AMP inspections by the general licensee, including trending of changes in identified inspection results. • Entries in the AMID for related dry storage system OE. • NRC documents such as Information Notices, Generic Letters or License Event Reports LERs resulting from dry fuel storage issues. • Relevant results of other domestic and international nuclear and nonnuclear research, inspection results and OE.
2	T_0 + 10 yrs (1)	Evaluate additional information gained from the sources listed in Tollgate 1 along with any new relevant sources and perform a written assessment of the aggregate impact of the information, including results of Tollgate 1. 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 1 assessment.
3	T_0 + 20 yrs (1)	Same as Tollgate 1 as informed by the results of Tollgates 1 and 2
4	T_0 + 30 yrs (1)	Same as Tollgate 1 as informed by the results of Tollgates 1, , 2 and 3.

Note: (1) T_0 is 20 years after MPC dry storage system began STORAGE OPERATIONS

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14.6 References

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

14.6.2 Fatigue Evaluation of NAC-MPC and UMS Storage System Components for Extended Storage, NAC-30013-2001, Revision 0

14.6.3 Time-Limited Aging Analysis (TLAA) for Potential Corrosion of the Steel Components in the YANKEE-MPC, CY-MPC AND LACBWR-MPC Storage System VCC Assembly for a Service Life of 60-Year, NAC-30013-2003, Revision 2

14.6.4 Aging Analysis for MPC-UMS Neutron Absorber and Neutron Shield Components (Storage -Transfer), NAC-30013-5001, Revision 0

14.6.5 Fuel Retrievability in Spent Fuel Storage Applications, ISG-2, Revision 2, April 26, 2016

14.6.6 NRC Certificate of Compliance for NAC-STC Transport Cask, Docket 71-9235, CoC No. 9253, Revision 19, November 2018

14.6.7 NEI 14-03, “Guidance for Operations Based Aging Management for Dry Cask Storage,” Revision 2, December 2016.

14.6.8 NAC International Submittal, NAC-MPC CoC Renewal Application, dated December 2019.

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

14.6.10 EPRI Technical Report, TR-3002008193, Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless-Steel Canisters

14.6.11 ED20170040, NAC Memorandum, Technical Report: NAC-UMS and NAC-MPC ISFSI and Individual TSC Rankings Based on EPRI CISCC Criteria,

14.6.12 EPRI Technical Report, TR-3002005371, Susceptibility Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless-Steel Canisters for Dry Storage

14.6.13 ACI 349.3R-18, “Report on Evaluation and Repair of Existing Nuclear Safety-Related Concrete Structures.” American Concrete Institute. 2018.

14.6.14 ANSI N14.6, “Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500kg) or More for Nuclear Materials”

14.6.15 ACI 349.3R-02, “Evaluation of Existing Nuclear Safety Related Concrete Structures (Reapp 2010)”

14.6.16 ASME Code Case N-860, “Inspection Requirements and Evaluation Standards for Spent Nuclear Fuel Storage and Transport Containment Systems,” July 2020.

14.6.17 ASME Code, Section XI, Division 1, Subsection IWE, 2007