Attachment 2

Request for Additional Information Responses

NAC-UMS 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-UMS CASK SYSTEM

(Docket No. 72-1015)

NAC International

February 2022

Section A

NON-PROPRIETARY RESPONSES

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

RAI 2-1:

Provide additional information to justify the exclusion of the transfer cask lead wool from the scope of the aging management review.

Revision 11 of the NAC-UMS FSAR, Table 2.3-1, stated that the lead wool in the transfer cask has operations and shielding functions. That entry has since been revised to remove the shielding function. Clarify the basis for that change to support the conclusion that the lead wool is not within the scope of renewal.

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

NAC International Response to RAI 2-1:

In accordance with Note 17 on NAC License Drawing No. 790-560 "Commercial grade lead wool may be used to fill open spaces by the trunnions as best possible." Item 23 on the drawing identifies lead wool as commercial grade material and it is defined in the FSAR as an NQ item. As the lead wool is only installed to fill gaps and does not perform a significant shielding function it has not been included in the scope of aging management review. In Chapter 5 of the FSAR, no shielding credit is identified for the lead wool in the shielding analysis. As there ae no identified loss of function for the lead wool and no identified degradation mechanisms identified for the lead bricks providing the most significant shielding function for the Transfer Cask, no further actions are required.

Summary of Changes:

No changes were made to the renewal application in response to this RAI.

NAC INTERNATIONAL RESPONSE
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RAI 3-1:

Provide additional justification for the conclusion that galvanic corrosion of the Vertical Concrete Cask (VCC) steel lid is not an aging effect requiring management.

Table 3.2-1 of the renewal application concludes that the VCC steel lid exposed to an outdoor air environment is not subject to galvanic corrosion. Section 3.2.1.1.3 of the renewal application provides an evaluation of galvanic corrosion of steel components; however, the evaluation did not include the steel lid in contact with the stainless steel lid bolts.

The conclusion that galvanic corrosion is not a credible aging mechanism does not appear to be supported by the pre-application inspection described in Appendix E. The inspection of the Maine Yankee NAC-UMS system found that the carbon steel lid bolt holes experienced corrosion due to interaction with the stainless steel lid bolts. That finding is also consistent with the NUREG-2214 conclusion that the galvanic corrosion of carbon steel is credible when in dissimilar metal contact in outdoor environments.

If galvanic corrosion is reconsidered to be a credible aging mechanism, revise the aging management review, as appropriate.

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

NAC International Response to RAI 3-1:

NAC has reviewed the applicable drawings for the NAC-UMS VCC and Transfer Casks and Transfer Adapters and has identified other components potentially exposed to galvanic corrosion in a sheltered and air-outdoor environment. The identified components are the stainless steel VCC lid bolts and washers and coated carbon steel VCC lid and top flange, and the stainless TFR retaining ring bolting and coated carbon steel retaining ring and top plate. Application for Renewal Section 3.2.1.1.3 has been revised as provided below. In addition, Application for Renewal Tables 3.2.2 and 3.2.3 have been revised to identify galvanic corrosion as a potential aging mechanism for the carbon and stainless-steel components. As the current Aging Management Programs 3 and 5 incorporate inspections of these components for indications of corrosion, loss of base material and loss of preload, no additional changes to the currents AMPs are required.

Summary of Changes:

Revised the NAC-UMS CoC Application as follows:

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-UMS 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 and the top of the steel VCC baffle weldment base plate. However, the potential for galvanic corrosion of the TSC stainless steel bottom plate is precluded by the presence of the 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 presence of the epoxy coating and the thickness of the baffle weldment top plate (2 inch).

Other points of connection between SSC steel subcomponents and more noble materials include 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 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 and lifting lug bolting, as applicable.

Another area of potential galvanic corrosion is at the interface between the Transfer Cask steel retaining ring and the stainless-steel (ferritic) retaining ring bolting. 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.

There are no other potential areas of galvanic corrosion identified for the NAC-UMS System. As galvanic corrosion rates of the VCC lid bolts and washers and retaining ring may be sufficient to affect component intended functions, galvanic corrosion is credible, aging management is required during the 40-year period of extended operation. The applicable AMP proposed to evaluate this aging mechanism is AMP-3 for the VCC components involved and the Transfer Cask AMP-5. These are discussed in Section 3.4.

Revisions to Tables 3.2.2 and 3.2.3

Tables 3.2.2 and 3.2.3 regarding VCC and TFR components were revised to address Galvanic Corrosion and to reference the applicable AMP.

NAC INTERNATIONAL RESPONSE
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RAI 3-2:

Provide a technical justification for the conclusion that thermal aging is not a credible aging mechanism for the precipitation-hardened stainless steel fuel basket support discs within the transportable storage canister.

Table 3.2-1 of the renewal application states that thermal aging of the 17-4 PH precipitation-hardened stainless steel fuel basket support discs is not a credible aging mechanism. In support of that conclusion, Section 3.2.1.2.8 of the renewal application states that the maximum long-term service temperature of the support discs is 316°C (601°F), which is below the 343°C (650°F) maximum allowable service temperature in ASME BPVC Section II, Part D.

As described in Section 3.2.2.8 of NUREG-2214, embrittlement of precipitation-hardened stainless steel has been observed below the ASME maximum-allowable temperature. NUREG-2214 cites a study by Olender et. al (2015), which provides operating experience and guidance for assessing the potential for embrittlement for operating temperatures between 243 and 316°C [470 to 600°F].

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

Reference

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.

NAC International Response to RAI 3-2:

As noted in Section 3.2.2.8 of NUREG-2214 that a review of thermal aging effects should be performed on a case-by-case basis for subcomponents constructed from Type 17-4PH stainless steel. Based on the discussions in the referenced study by Olender et. al, the applicable conditions for the 17-4PH support disks of the NAC-UMS PWR basket are evaluated, which include initial heat treatment condition, service temperature, operating environment, and stress level. As shown in the following discussion, thermal aging will not compromise the safety function of the 17-4PH support disks, based on some of the recommended actions in the Olender et. al. study. (Note: Some of the recommended actions would only apply to the design phase and are not addressed in this case study analysis.)

Material Condition

As noted in licensing drawing 790-593, the support disks are made of 17-4PH stainless steel per ASME SA-693, Type 630, heat treated to condition H1150. In the recommended actions in the referenced study by Olender et. al, the H1150 is considered optimally heat treated, if the design allows it, which is the case for this application.

Service Temperature

The study by Olender et. al provides guidance for assessing the potential for embrittlement for the 17-4PH stainless steel for operating temperatures between 470°F and 600°F. The maximum temperature of 601°F of the 17-4PH support disks corresponds to the design basis heat load case for the normal condition of storage. This temperature occurs at a very limited region at the center of the middle disk of the basket. The average temperature of all the support disks is 396°F, with temperature at the center of the disks ranging from 294°F to 601°F and the temperature at the disk edges ranging from 165°F to 400°F. Note that this temperature profile corresponds to the design basis heat load condition. Also note that the operating temperatures for all the system components continually decrease during the lifetime of the storage as a result of fuel decay.

Operating Environment

Most cases of the failures of 17-4PH components observed and described in the study by Olender et. al occurred in active components such as valves or subcomponents of valves, which were in a pressure boundary and were subjected to dynamic loading. This indicates that the failures are associated with large primary stresses. Additionally, much of the operating experiences described in the study indicate an element of corrosion, SCC or overloading contributing to the observed failure.

During the long-term storage condition, the basket support disks maintain the locations of each fuel tube for criticality control. The support disks are individually supported at eight tie-rod locations and are subjected to static load from the self-weight of the disks only. The support disks are not subjected to dynamic or cyclic loads.

Since the canisters are backfilled with helium, the support disks are in an inert environment, not an aggressive chemical environment as discussed in the study by Olender at. el. Also, much of this operating experience in the study is in a system pressure boundary or subject to dynamic loads on an active component. The loading condition and the operating environment for the support disks in the long-term storage conditions are static and mild. This type of operating environment would not challenge the safety function of the support disks exposed to any potential thermal aging effects.

Stress Level

During the long-term storage condition, the support disks are in a horizontal position and individually supported at eight tie-rod locations. Each support disk is subjected to a static inertial load corresponding to its self-weight only, resulting in minimal stresses in the disk. As shown in Table 3.4.4.1-12 of the NAC-UMS FSAR, the maximum stress intensity for the primary membrane plus primary bending ($P_m + P_b$) stress is 0.8 ksi. Using the allowable stress as defined by the ASME Code Subsection NG is 52.7 ksi, which is based on a conservative temperature of 800°F, the minimum margin of safety for the support disks is 64.8. Since the support disks are under such a low stress level, any thermal aging effect on this favorably heat-treated material would have a negligible effect on the structural function of the support disk.

Conclusion of Evaluation

The 17-4PH support disks for NAC-UMS PWR system are in a service where some portion of them can be exposed to operating temperatures between 470°F and 600°F, which can potentially cause embrittlement. However, the component failures described in the study by Olender at. el. are associated with the combined conditions of high stresses due to dynamic or cyclic loads and elevated operating temperatures. Therefore, based on the discussion above, any thermal aging effect will not adversely impact the safety function of the 17-4PH support disks because of the following characteristics and conditions.

- The support disks have been heat treated to an optimum condition to minimize the susceptibility of the material to thermal embrittlement.
- Only a limited portion of the disks are subjected to temperatures above 470°F and the average disk temperature is below 400°F for normal conditions of storage (Note that all disks are subjected to negligible static primary stresses).
- The disks are stainless steel materials in an inert operating environment. The loading condition and the operating environment for the support disks in the long-term storage conditions are static and mild.
- The support disks are subjected to insignificant static loads (self-weight only) with a very low stress level.

Summary of Changes:

The above discussion is documented in Appendix A of NAC Calculation No. 30013-2001 Rev. 2. Appendix B of the renewal package (Time-Limited Aging Analysis) will be updated to include Rev. 2 of this calculation. Sections 3.2.1.2.8 and 3.3.3.1, and Table 3.2-1 of the renewal application are revised to indicate that thermal aging of the 17-4 PH stainless steel fuel basket support disks is dispositioned by TLAA. Chapter 14 Section 14.2 is revised to include a new section (14.2.4) for the TLAA for the thermal aging of the 17-4PH stainless steel support disks.

NAC INTERNATIONAL RESPONSE
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Appendix A - Aging Management Programs
RAI A-1:

Provide a justification for the option in the Localized Corrosion and Stress Corrosion Cracking of Welded Stainless-Steel Transportable Storage Canisters (TSC) aging management program (AMP) that allows a general licensee to choose not to conduct the canister inspections. Otherwise, revise the renewal application to remove that option.

The AMP for the canister inspections provides an ISFSI site the option of not conducting inspections if the site provides a justification. The AMP does not provide any criteria for that justification. In addition, the staff notes that, if the site does not perform the canister inspections, the site would also not perform the activities under the Internal VCC Metallic Components Monitoring AMP. The inspections under the VCC AMP occur only when canister inspections are conducted.

Provide additional information that justifies this approach, addressing the following:

- The TSC AMP provides the option for not conducting inspections in a manner that does not appear to require a general licensee to evaluate such a deviation under the provisions of 10 CFR Parts 72.212 and 72.48 (see RIS 2012-05). The staff notes that the guidance in NUREG-1927, Appendix E, and NEI 14-03, Section 2.2.3, cite the use of the 72.212 and 72.48 processes as the appropriate means to make changes to aging management programs. As articulated in NEI 14-03:

10 CFR 72.48 provides the appropriate set of public health and safety-based criteria for determining whether NRC review and approval of revised TLAAs and AMPs is required prior to implementation.

NAC International Response to RAI A-1:

NAC has deleted this option from Element 4 of Table A-1, AMP-1 "Aging Management Program for Localized Corrosion and Stress Corrosion Cracking of Welded Stainless-Steel Transportable Storage Canisters." The revised AMP is included in the FSAR Chapter 14 and in the CoC Renewal Application.

Summary of Changes:

Deleted the following paragraph from Element 4 of AMP-1, "Justification for not conducting inspections for localized corrosion or SCC will be provided on a case-by-case basis for each ISFSI site where welded TSCs are in use."

NAC INTERNATIONAL RESPONSE
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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 <u>AMP-1 Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSCs)</u>

In lieu of utilizing the proposed inspection guidelines and acceptance criteria proposed in Table 6.2 of NUREG-2214, NAC-UMS 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 (3.9.370) for supplemental examination of major indications as documented in the proposed AMP. To support identification of the most susceptible TSCs to SCC, all NAC-UMS 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 (3.9.370). 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."

NAC INTERNATIONAL RESPONSE
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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-UMS 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.

NAC INTERNATIONAL RESPONSE
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Appendix A - Aging Management Programs
RAI A-4:

Clarify the basis for the conclusion that visual inspections will be capable of verifying the shielding performance of concrete in the Reinforced VCC Structures – Concrete Monitoring AMP.

Section 3.4.3.4 of the renewal application states that shielding tests of the VCC concrete are not needed, citing an NRC analysis of the use of ACI 349.3R visual inspection parameters to evaluate concrete shielding efficacy (NRC, 2019). However, the cited NRC analysis did not include an evaluation of the NAC-UMS system. In addition, the analysis found that the comparable NAC-MPC and similar systems with BWR contents may not be able to rely on the ACI 349.3R inspection parameters to assess shielding performance. Given this conclusion of the cited analysis, provide additional information to support the conclusion that visual inspections, in lieu of radiation surveys, can be used to ensure the VCCs maintain their shielding function in the period of extended operation.

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

Reference

NRC. "Study of ACI 349.3R Concrete Evaluation Criteria Impacts on Dose Rates for Several Spent Fuel Dry Storage System Designs." Washington, DC. ADAMS Accession No. ML19072A031. 2019

NAC International Response to RAI A-4:

AMP-4 has been revised to require a gamma shielding test in the areas of any new Tier 3 cracks or indications identified during the VCC inspection. The gamma shielding results will be evaluated against the LCO A 3.2.2 dose rate limits to determine if further corrective actions are required.

Summary of Changes:

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

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.
- Increase in Gamma Dose Rates exceeding LCO A 3.2.2 levels.

AMP-4 Element 6 has been revised to read as follows:

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 new tier 2 findings may be accepted after review by the designated responsible-in-charge engineer.
- All new tier 3 findings must be reviewed by the designated responsible-in-charge engineer and are subject to further evaluations as appropriate for the finding. New tier 3 indications or any other indication which could potentially increase dose rates shall be subject to Gamma Dose rate measurements and verified to be less than LCO A 3.2.2 acceptance criteria.

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)
- Increases in Gamma Dose Rates exceeding LCO A 3.2.2 acceptance criteria

NAC INTERNATIONAL RESPONSE
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Appendix A - Aging Management Programs
RAI A-5:

For those facilities that do not normally maintain a Transfer Cask and Transfer Adapter on site, clarify the controls that will be in place to ensure that aging management activities will be performed prior to placing those components into service at the site.
Table A-5 and proposed FSAR Table 14.3-9 in the appendices of the renewal application state that the Transfer Cask/Transfer Adaptor AMP is not applicable to facilities not maintaining the components on site. It is not clear to the staff what controls will be in place to ensure that aging of the transfer components will be managed. The AMP statements presume that aging management activities will be performed by another facility housing these components. However, given the fact that the AMP does not require periodic inspections for transfer casks that are not in use for extended periods (but rather requires a return-to-service inspection in that case), a general licensee cannot assume that aging management inspections have been performed by the facility normally housing the transfer cask.

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

NAC International Response to RAI A-5:

It was always the intent of NAC that any Transfer Cask/Transfer Adapter provided to NAC-UMS Licensees will be required to meet the AMP-5 requirements, or the applicable requirements of the Transfer Cask Procurement Specification, which is based on ANSI N14.6 requirements.

Summary of Changes:

FSAR Chapter 1, Section 1.2.1 has been revised to add the following:

For decommissioned sites that have disposed of their auxiliary equipment, new or refurbished equipment shall be used to allow the successful transfer of the NAC-UMS Transportable Storage Canisters (TSCs) from the VCC to the appropriate NAC-UMS System transport cask certified for the transport of the NAC-UMS TSCs. New transfer equipment including the Transfer Cask and Transfer Adapter shall be procured, inspected, and tested in accordance with approved NAC Procurement Specification and applicable License Drawings. If currently existing equipment is to be utilized, the refurbished equipment will be required to comply with all requirements of the applicable Transfer Cask and Transfer Adapter Aging Management Program, as will equipment that is in storage at some of the decommissioned sites prior to delivery to the NAC-UMS Licensees.

In addition, Element 1 of AMP-5 has been revised as follows:

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.

NAC INTERNATIONAL RESPONSE
TO
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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 (3.9.372) for examination requirements including IWE-2311 General Visual Examination, IWE-2330(b) Personnel Qualification, and IWE-3511 General Visual Examination of Coated and Uncoated Areas Acceptance Standards or their equivalent requirements.

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:

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, 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 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.

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.

**NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION**

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] 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 section of the FSAR Chapter 9 (9.2.1 and 9.2.1) have been revised to correct the proper start of AMP implantation.

Summary of Changes:

FSAR Section 9.2.1 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-UMS 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 9.2.1.

FSAR Section 9.2.2 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-UMS 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 9.2.2.

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ENCLOSURE 1

APPLICATION FOR RENEWAL OF THE NAC-UMS SYSTEM CoC

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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
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
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

1.0 GENERAL INFORMATION

The NAC-UMS Universal Storage System (hereafter referred to as the NAC-UMS System) is approved under 10 CFR 72, Subpart K (Docket No. 72-1015) 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-UMS System Certificate of Compliance (CoC) No. 1015 was initially issued by the U.S. Nuclear Regulatory Commission (NRC) on November 20, 2000 with an expiration date of November 20, 2020. NAC International (NAC), as the Certificate Holder (CH) of the NAC-UMS System CoC No. 1015 [1.3.1.a. through 1.3.1.h.], is applying for renewal of CoC No. 1015 for a term of 40 years in accordance with 10 CFR 72.240(a).

Additionally, NAC is applying for renewal of the initial NAC-UMS System CoC and Amendments 1 through 7.

The requested 40-year CoC renewal term will extend the CoC expiration date to November 20, 2060. The NAC-UMS 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 (UFSAR) [1.3.2] 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 (SSC) Important-to-Safety (ITS).

In accordance with 10 CFR 72.240(d), the NAC-UMS 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-UMS CoC and Amendment History

The initial NAC-UMS System CoC No. 1015 [1.3.1.a.] was issued on November 20, 2000 based on NAC-UMS Safety Analysis Report (SAR), Revision 4. The original CoC approved the NAC-UMS System designed for the storage of five classes of Transportable Storage Canisters (TSCs) including three lengths for PWR fuel types and two lengths for BWR fuel types. The system included a TSC provided with integral fuel baskets for the storage of up to 24 PWR and 56 BWR spent fuel assemblies. Subsequently, seven (7) amendments were issued to the NAC-UMS System CoC. A summary of the NAC-UMS 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 July 16, 1999, as supplemented on October 20 and November 16, 1999, and February 4 and 7, March 17, April 18, May 31, June 19, July 27, August 24, and September 1, 6 and 12, 2000, NAC requested NRC approval of an amendment to CoC No. 1015 for the NAC-UMS System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. The proposed amendment requested: (1) changes to authorized contents to allow Maine Yankee site-specific spent fuel assemblies within the PWR basket including damaged or consolidated fuel in a Maine Yankee damaged fuel can (DFC), and fuel burnups of up to 50,000 MWd/MTU; (2) changes to allow longer times for PWR spent fuel cask loading operations based on reduced heat loads; (3) authorization to store without canning intact PWR fuel assemblies with missing grid spacers (up to an unsupported length of 60 inches); (4) editorial clarifications to the Technical Specifications (TSs); and (5) deletion of a reference to the NS-4-FR trade name of the solid neutron shielding for the VCC shield plug. The request, as supplemented, was approved by the NRC in Amendment No. 1 [1.3.1.b.] and was effective February 20, 2001.

- **Amendment No. 2:** By application dated October 17, 2000, as supplemented on December 7, 2000, April 27, July 5, July 18, July 19, July 26, and August 1, 2001, NAC requested NRC approval of an amendment to CoC No. 1015 for the NAC-UMS System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. The proposed amendment requested: (1) changes to authorized contents to allow Maine Yankee to store various components associated with spent fuel assemblies; (2) a Technical Specification (TS) change to delete the requirement to place a TSC in the transfer cask if the associated VCC's vents cannot be unblocked; and (3) editorial clarifications, administrative changes and correction of discrepancies in the Technical Specifications (TSs). The request, as supplemented, was approved by the NRC in Amendment No. 2 [1.3.1.c.] and was effective December 31, 2001

- **Amendment No. 3:** By application dated January 15, 2002, as supplemented on February 4, July 3, August 7, November 27, and December 11, 2002, and August 15, 2003, NAC requested NRC approval of an amendment to CoC No. 1015 for the NAC-UMS System in accordance with the provisions of 10 CFR Part 72, Subparts K and L to incorporate Enhanced Design Features (E-UMS) to the NAC-UMS System. The following changes were incorporated: (1) adding METAMIC as an alternative neutron poison material; (2) revising the structural analysis to address an increase in the boiling water reactor (BWR) fuel assembly weight from 696 to 702 pounds, an upgrade in the design earthquake levels, and modification to the transfer cask trunnion support design; (3) revising the fuel assembly dimensions (length and width) for more comprehensive coverage of BWR and PWR fuel types; (4) revising the thermal analyses to extend operating time limits for vacuum drying, canister in transfer cask, helium backfill, forced air cooling, and in-pool cooling; (5) revising allowable fuel cladding temperatures to reflect "Cladding Considerations for the Transport and Storage of Spent Fuel," Interim Staff Guidance – 11, R2 (ISG-11); (6) incorporating criticality analyses for loading of 5.0 weight % U235 enriched PWR fuel with a minimum of 1000 ppm soluble boron; (7) reorganizing the Criticality Section to separately describe SCALE and MONK computer models; (8) revising the Technical Specifications to delete annual effluent reporting requirements; and (9) incorporating editorial and administrative changes to the CoC and Technical Specifications (TSs), including revision to the CoC format. The request, as supplemented, was approved by the NRC in Amendment No. 3 [1.3.1.d.] and was effective March 31, 2004.

- **Amendment No. 4:** By application dated August 10, 2004, as supplemented on December 23, 2004, and February 17, 2005, NAC requested NRC approval of an amendment to CoC No. 1015 for the NAC-UMS System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. The proposed amendment requested the following changes: (1) to replace the specific term "zircaloy" with the more generic term "zirconium alloy"; (2) to revise the Technical Specification definitions of OPERABLE and SITE SPECIFIC FUEL; (3) to revise Technical Specification (TS) vacuum drying pressure and time limits; (4) to revise the REQUIRED ACTIONS and COMPLETION TIMES of the Technical Specification LCO for CONCRETE CASK Heat Removal System; (5) to clarify the Technical Specification LCO for CONCRETE CASK Average Surface Dose Rate SURVEILLANCE FREQUENCY; (6) to add an option to Technical Specification LCO ACTIONS for Dissolved Boron Concentration for restoring the dissolved boron concentration; (7) to delete the redundant administrative control for boron concentration from the Technical Specification LCO for Dissolved Boron Concentration; (8) to revise the Technical Specification (TS) to add an alternate site-specific design basis earthquake (DBE) analysis for unbounded site conditions; and (9) to incorporate editorial and administrative changes. The request, as supplemented, was approved by the NRC in Amendment No. 4 [1.3.1.e.] and was effective October 11, 2005.

- **Amendment No. 5:** By application dated September 22, 2006, as supplemented on May 8, September 6, September 10, September 26, and November 30, 2007, and April 23 and May 8, 2008, NAC requested NRC approval of an amendment to CoC No. 1015 for the NAC-UMS System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. The proposed amendment would revise the CoC and TS for the NAC-UMS System, and revision of the FSAR, to allow for storage of high burnup PWR fuel (up to 60 GWd/MTU assembly average burnup). The other proposed amendment changes were: (1) elimination of the requirement for the use of a tamper-indicating seal on the concrete cask lid bolts; (2) elimination of the Technical Specification requirement (Limiting Condition for Operation [LCO] A 3.1.5) for the helium leakage test of the canister shield lid to canister shell weld; (3) revision of the Technical Specification, Section B2.1 written reporting requirement to 60 days (was 30 days); (4) elimination of the requirement for Charpy V-notch impact testing of the 0.625-inch nominal thickness BWR support disk material (SA 533, Type B, Class 2); and (5) revised Delta Note 8 on License Drawing 790-585 to make the use of the structural lid and shield lid threaded plugs and dowel pins optional. In its September 6, 2007, response to the Staff's request for additional information (RAI) dated May 30, 2007, NAC supplemented its application to: (1) reinstate the shield lid to canister shell weld helium leak testing requirements; (2) revise the definitions of intact and damaged fuel to conform with the NRC guidance of Interim Staff Guidance (ISG) ISG-1, Revision 2; and (3) revise the TS requirements to limit potential thermal cycling of the fuel cladding to conform with the staff's position in ISG-11, Revision 3. The request, as supplemented, was approved by the NRC in Amendment No. 5 [1.3.1.f.] and was effective January 12, 2009.

- **Amendment No. 6:** By application dated May 23, 2017, as supplemented on January 16, 2018, NAC requested NRC approval of an amendment to CoC No. 1015 for the NAC-UMS System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. The proposed amendment would revise the NAC-UMS CoC to change Technical Specification No. A.3.1.6, “CONCRETE CASK Heat Removal System,” clarify the applicability of Technical Specification No. A.3.2.2, “Concrete Cask Average Surface Dose Rates,” delete Technical Specification No. A.5.4, “Surveillance After an Off-Normal, Accident, or Natural Phenomena Event,” and revise the FSAR, Chapter 12, which provides the basis for Limiting Condition of Operation (LCO) C 3.1.6 to provide additional guidance for the intent of “immediate” actions. The amendment request, as supplemented, was approved by the NRC in Amendment No. 6 [1.3.1.g.] and was effective January 7, 2019.

- **Amendment No. 7:** By application dated September 18, 2018, NAC requested NRC approval of an amendment to CoC No. 1015 for the NAC-UMS System in accordance with the provisions of 10 CFR Part 72, Subparts K and L. The proposed amendment would revise the NAC-UMS FSAR, Chapter 12, to clarify the Technical Specification surveillance requirements bases for Limiting Condition of Operation (LCO) C 3.1.6.1 to state that "A minimum of two outlet air temperatures must be measured to provide an average outlet air temperature measurement to comply with Technical Specification surveillance requirement 3.1.6.1." The amendment request was approved by the NRC in Amendment No. 7 [1.3.1.h.] and was effective July 29, 2019.

1.1.2 NAC-UMS Storage System Loading Overview

General

Through the end of 2017 a total of two hundred sixty four (264) NAC-UMS Systems were loaded with SNF at four Independent Spent Fuel Storage Installations (ISFSIs): sixty (60) systems at Maine Yankee Atomic Power Company's (MYAPC) Maine Yankee (MY) ISFSI in Maine; twenty eight (28) systems at Duke Energy's McGuire Nuclear Station (MNS) in North Carolina; twenty four (24) systems at Duke Energy's Catawba Nuclear Station (CNS) in South Carolina; and a total of one hundred and fifty two (152) systems at Arizona Public Service's (APS) Palo Verde Nuclear Generating Station (PVNGS) in Arizona. No additional loading operations were performed after September 1, 2017, and no additional NAC-UMS Systems are planned for loading at the time of this CoC renewal request.

Maine Yankee NAC-UMS System Loading Operations

NAC-UMS System loading operations began at the shutdown MY power plant with the first system placed into service on August 24, 2002 and the last system placed into service on February 27, 2004. The lowest heat load system placed into service was fuel loading operation #4 at 4.31 kW on September 21, 2002, and the highest was #47 at 13.95 kW on December 6, 2003. The maximum fuel burnup loaded for the 14 x 14 CE PWR SFAs was 49,241 MWd/MTU. A total of 137 Damaged Fuel Cans (DFCs) were loaded into thirty-eight (38) MY NAC-UMS Systems including: 90 fuel assemblies with burnups of > 45,000 MWd/MTU; 43 SFAs with defects requiring canning per the Technical Specifications; 2 Damaged Fuel Rod Holders; and 2 consolidated fuel assemblies.

As identified in Table 1.1-1 below, all MY NAC-UMS Systems including Transportable Storage Canisters (TSC), Vertical Concrete Casks (VCC), and Damaged Fuel Cans (DFC) [DFC (L) and Fuel Only Cans (FOC)] were fabricated, delivered and loaded under Amendment No. 2 of the CoC. This includes UMS-TSC-790-001 through -060, UMS-VCC-790-001 through -060, UMS-DFC (L) 790-001 through -060 and UMS-DFC (FOC) 790-001 through -078.

Following completion of the MY fuel transfer operations campaign and at Maine Yankee's request, NAC completed NAC Calculation No. 12412-9000, Revision 0, dated January 15, 2010 [1.3.3], in which a review was performed to confirm that the MY NAC-UMS TSCs, VCCs and DFCs fabricated, constructed, loaded and certified to CoC Amendment No. 2, could be recertified to CoC Amendment No. 5 and FSAR Revision 9. The analysis and review confirmed that the MY NAC-UMS system hardware, fuel contents and loading operations could be recertified to CoC Amendment No. 5. In the case of loading and transfer of UMS-TSC-790-016, which although compliant with the TS conditions of Amendment No. 2 exceeded the allowable time in Transfer Cask of 25 days implemented in CoC Amendment No. 3 and which remains in effect through CoC Amendment No. 5, Maine Yankee requested and was granted an exemption [1.3.5 and 1.3.6]. As a result of the reconciliation review, a Supplemental Certificate of Conformance MY-COC-TSC-VCC-DFC [1.3.4] was issued by NAC International on January 22, 2010 recertifying the MY NAC-UMS hardware to NRC CoC No. 1015, Amendment No. 5 and FSAR Revision 9. Maine Yankee notified the NRC of the recertification of the MY NAC-UMS Systems to CoC 1015, Amendment 5 and UFSAR Revision 9 [1.3.7] in accordance with 10 CFR 72.212(b)(4).

Following the NRC approval of Amendment 6 to the NAC-UMS CoC, MY has recertified all their NAC-UMS Systems to Amendment 6 as indicated in Table 1.1-1 below. Maine Yankee notified the NRC of the recertification of the MY NAC-UMS Systems to CoC 1015, Amendment 6 and UFSAR Revision 13 [1.3.21] in accordance with 10 CFR 72.212(b)(4). No further NAC-UMS System loadings are foreseen at MY.

Table 1.1-1 Maine Yankee NAC-UMS System Components CoC Compliance Matrix

Maine Yankee System Number	Registered Amendment Usage by the Licensee per 10 CFR 72.212(b)(2)			
	TSC Fabrication	VCC Fabrication	DFC Fabrication	LOADING
NAC-UMS 1-60	Amendment 6	Amendment 6		Amendment 6
DFC(L) 1-60			Amendment 6	
DFC (FOC) 1-78			Amendment 6	
Maine Yankee System Number	**NAC-UMS CoC Original as Fabricated Amendment**			
	TSC Fabrication	**VCC Fabrication**	**DFC Fabrication**	**LOADING**
NAC-UMS 1-60	Amendment 2	Amendment 2		Amendment 2
DFC(L) 1-60			Amendment 2	
DFC (FOC) 1-78			Amendment 2	
	NAC-UMS Transfer Cask (1) Fabrication			
Unit No. 1	N/A			
Unit No. 2	N/A			

(1) After placing all its spent nuclear fuel into dry storage, MY sold one of its NAC-UMS Transfer Casks (TFR) to Duke Energy Corporation. Duke now owns and maintains one MY NAC-UMS TFR and there was no reconciliation analysis performed for MY regarding this transfer system. MY returned the second NAC-UMS TFR to NAC at the end of the campaign and site decommissioning. NAC has taken the second TFR out of service and currently stores the TFR with other NAC equipment at Alaron Corporation's licensed storage facility in Wampum, PA

APS's PVNGS NAC-UMS System Loading Operations

NAC-UMS System loading operations began in at PVNGS with the first system placed into service March 15, 2003 and the last system (#152) was placed into service at the ISFSI on September 1, 2017. The lowest heat load system was fuel loading #10 and the highest heat load was fuel loading # 117 on July 18, 2014. The maximum 16 x 16 CE PWR SFAs fuel burnup loaded in a NAC-UMS TSC was 54,553 MWd/MTU and was during loading operation # 117 on July 18, 2014 (additional SFAs of identical burnup were loaded during fuel loading operations # 118, 119, and 120 through August 8, 2014). A total of 1,156 high burnup (HBU) fuel assemblies were loaded beginning with TSC fuel load #59 in March 2009 following approval of NAC-UMS CoC No.1015 Amendment 5. All the HBU assemblies were loaded into a total of 93 NAC-UMS TSCs without canning in Damaged Fuel Cans (DFCs). Only undamaged SFAs have been loaded into NAC-UMS Systems at PVNGS.

As shown in Table 1.1-2 below the APS PVNGS NAC-UMS units were initially fabricated, constructed, and loaded under NRC CoC No. 1015 Amendments 3, 4 and 5. At APS's request following the loading of UMS System No. 84, a review was performed to confirm that the APS PVNGS NAC-UMS TSCs and VCCs fabricated, constructed, loaded and certified to

CoC Amendment Nos. 3 and 4 could be recertified to CoC Amendment No. 5 and UFSAR in effect at time of system fabrication, fabrication and loading. The reconciliation review was documented in NAC International, Inc., Calculation No. 12407-2005, R0, Arizona Public Service Palo Verde Nuclear Generating Station, "NAC-UMS Certificates of Compliance Reconciliation of Fabrication & Construction and Fuel Content Limitations for UMS-PWR Transportable Storage Canisters, Vertical Concrete Casks, and Transfer Casks," dated January 8, 2010 [1.3.8]. The analysis and review confirmed that the APS NAC-UMS system hardware, loading operations and fuel contents can be recertified to CoC Amendment No. 5. As a result of the reconciliation review, Supplemental Certificate of Conformance PV-COC-TSC/VCC 1-84 [1.3.9] and Supplemental Certificate of Conformance PV-COC-TFR [1.3.10] were issued by NAC International on January 14, 2010 recertifying the PVNGS NAC-UMS System hardware to NRC CoC No. 1015, Amendment No. 5 and UFSAR Revision 9. All NAC-UMS Systems loaded after the completion of the reconciliation after Unit # 84 have been fabricated, constructed and loaded in compliance with NRC CoC No. 1015, Amendment No. 5 and FSAR in effect at time of system construction, fabrication, and loading.

APS notified the NRC of the recertification of the PVNGS NAC-UMS Systems to CoC 1015, Amendment 5 and UFSAR Revision 9 [1.3.11] in accordance with 10 CFR 72.212(b)(4).

Table 1.1-2 APS PVNGS NAC-UMS Components CoC Compliance Matrix

PVNGS NAC-UMS System Number	Registered Amendment Usage by the Licensee per 10 CFR 72.212(b)(2)		
	TSC Fabrication	VCC Fabrication	LOADING
1-152	Amendment 5	Amendment 5	Amendment 5
	NAC-UMS Advanced Transfer Cask		
Unit #1 790-060-95	Amendment 5		
PVNGS NAC-UMS System Number	NAC-UMS CoC Original as Fabricated Amendment		
	TSC Fabrication	VCC Fabrication	LOADING
1-2	Amendment 4	Amendment 3	Amendment 4
19-62	Amendment 4	Amendment 4	Amendment 4
40-41	Amendment 4	Amendment 3	Amendment 4
42-58	Amendment 4	Amendment 4	Amendment 4
59-62	Amendment 4	Amendment 4	Amendment 5
63	Amendment 4	Amendment 5	Amendment 5
64-66	Amendment 4	Amendment 4	Amendment 5
67-68	Amendment 5	Amendment 4	Amendment 5
69-152	Amendment 5	Amendment 5	Amendment 5
	NAC-UMS Advanced Transfer Cask Fabrication		
Unit #1 790-060-95	Amendment 2		

No further NAC-UMS System loading operations are foreseen at APS's PVNGS. PVNGS is currently loading NAC MAGNASTOR Systems and final number of MAGNASTOR Systems to be loaded at PVNGS is unknown at current time

<u>Duke Energy's McGuire and Catawba Nuclear Stations NAC-UMS System Loading Operations</u>

NAC-UMS System loading operations began at Duke Energy's McGuire Nuclear Station (MNS) with the first system placed into service on December 22, 2004 and the last system placed into service on May 8, 2009. The NAC-UMS Systems at MNS are co-located with ten TN-32 metal storage casks loaded prior to the start of loading operations of NAC-UMS Systems at MNS. After the end of NAC-UMS System loading operations, MNS transitioned to loading newer NAC MAGNASTOR Systems, which began first loading operations in 2013 and are continuing. The final number of MAGNASTOR Systems to be loaded at MNS is unknown at the current time.

The lowest heat load NAC-UMS System placed into service was fuel loading operation #1 at 10.23 kW, and the highest was #27 at 16.75 kW on April 24, 2009. The maximum fuel burnup loaded for the 17 x 17 W PWR SFAs was 44,987 MWd/MTU. A total of twenty-eight (28) NAC-UMS Systems (Units 1-24 and 49-52) were loaded at MNS. Only undamaged SFAs have been loaded at MNS and no HBU assemblies were loaded into the MNS NAC-UMS Systems.

NAC-UMS System loading operations began at Duke Energy's Catawba Nuclear Station (CNS) with the first system placed into service on July 30, 2007 and the last system placed into service in December 2011. After the last NAC-UMS System was placed on the ISFSI, CNS began loading newer MAGNASTOR Systems in 2013, and MAGNASTOR loading operations are continuing. The final number of MAGNASTOR Systems to be loaded at CNS is unknown at the current time.

The lowest heat load NAC-UMS System placed into service at CNS was fuel loading operation #1 at 12.22 kW, and the highest was #14 at 17.58 kW on May 17, 2010. The maximum fuel burnup loaded for the 17 x 17 W PWR SFAs at CNS was 44,967 MWd/MTU in UMS-TSC-418-036. A total of twenty-four (24) NAC-UMS Systems (Units 25-48) were loaded at CNS. Only undamaged SFAs have been loaded at CNS and no high burnup assemblies were loaded into the CNS NAC-UMS Systems.

As shown in Table 1.1-3 below, the Duke Energy MNS and CNS NAC-UMS Systems were initially fabricated, constructed and loaded under NRC CoC No. 1015 Amendments 2, 3 and 4. At the request of Duke Energy, NAC International, Inc. performed two NRC CoC No. 1015 reconciliations in NAC Calculation No. 12419-2009, Duke Energy McGuire and Catawba Nuclear Stations, "NAC-UMS® Certificate of Compliance Amendment Reconciliation of Fabrication & Construction Activities, Operational Constraints and Fuel Contents Limitations UMS-PWR Transportable Storage Canisters, Vertical Concrete Casks & Transfer Casks" [1.3.12]. R0 was issued on July 3, 2007 reconciling Duke Energy TSC and VCC Units 1-48 and MNS and CNS Transfer Casks to NRC CoC No. 1015, Amendment 4. NAC Calculation No. 12419-2009, R1 was issued on January 12, 2010 to reconcile all Duke Energy hardware, fuel contents and operations to NRC CoC No. 1015, Amendment 5. As a result of the Revision 1 to the calculation, NAC Certificate of Conformance (COC), Duke Energy, McGuire and Catawba Nuclear Stations, January 14, 2010 [1.3.13] was prepared and issued. In addition to this overall COC, NAC International also issued the following: Supplemental Certificate of Conformance, Duke Energy – McGuire Nuclear Stations, NAC-UMS Units 1-24, dated January 14, 2010 [1.3.14]; Supplemental Certificate of Conformance, Duke Energy – McGuire Nuclear Station, NAC-UMS Transfer Cask, Serial No. 01-1235-01, January 14, 2010 [1.3.15]; Supplemental Certificate of Conformance, Duke Energy – Catawba Nuclear Stations, NAC-UMS Units 25-48, dated January 14, 2010 [1.3.16]; and Supplemental Certificate of Conformance, Maine Yankee Atomic Power Station (Original Owner) and Duke Energy-Catawba Nuclear Station (Current Owner), NAC-UMS Transfer Cask, Unit No. MY-790-066-99, January 14, 2010 [1.3.17]. However, Duke Energy does not intend to register the McGuire and Catawba NAC-UMS Systems as certified to CoC No. 1015 Amendment 5 and will maintain their current operation under Amendment 4

Table 1.1-3 Duke Energy MNS and CNS NAC-UMS Components CoC Compliance Matrix

Duke Energy NAC-UMS System Number	Registered Amendment Usage by the Licensee per 10 CFR 72.212(b)(2)		
	TSC Fabrication	**VCC Fabrication**	**LOADING**
MNS #'s 1-24, 49-52	Amendment 4	Amendment 4	Amendment 4
CNS #'s 25-48	Amendment 4	Amendment 4	Amendment 4
	NAC-UMS Transfer Cask		
McGuire	Amendment 4		
Catawba (Procured from MY)	Amendment 4		
Duke Energy NAC-UMS System Number	**NAC-UMS CoC Original as Fabricated Amendment**		
	TSC Fabrication	**VCC Fabrication**	**LOADING**
MNS #'s 1-16	Amendment 2	Amendment 2	Amendment 3 (#'s 1-5; Amendment 4 (#'s 6-16)
MNS #'s 17-24	Amendment 3	Amendment 3	Amendment 4
CNS #'s 25-36	Amendment 4	Amendment 3	Amendment 4
CNS #'s 37-48	Amendment 4	Amendment 4	Amendment 4
MNS #'s 49-52	Amendment 4	Amendment 4	Amendment 4
	NAC-UMS Transfer Cask Fabrication		
McGuire	Amendment 2		
Catawba (Procured from MY)	Amendment 1		

Currently, Duke Energy has registered all McGuire and Catawba Nuclear Stations' NAC-UMS units with the NRC as certified under NAC-UMS CoC No. 1015 under Amendment 4 and UFSAR Revision 7 [1.3.20].

No further NAC-UMS System loading operations are foreseen at Duke Energy's McGuire and Catawba Nuclear Stations. Both MNS and CNS are currently loading NAC MAGNASTOR Systems.

<u>Overall NAC-UMS Operational Experience</u>

No significant storage, loading, transfer, operational, off-normal or accident events have occurred at any of the four ISFSI facilities utilizing the NAC-UMS Systems. Lessons learned during initial loading operations at MY, PVNGS, MNS and CNS are discussed in Section 3.

1.2 APPLICATION FORMAT AND CONTENT

The NAC-UMS 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 [1.3.18], and NEI 14-03 [1.3.19]. Table 1.2-1 provides a summary of the section number and headings of the NAC-UMS System CoC renewal application and cross-references to the applicable sections of NUREG-1927 [1.3.18] and 10 CFR Part 72 Regulations. All changes in the NAC-UMS System that have been previously made without prior NRC approval in accordance with 10CFR72.48 have been incorporated into the latest UFSAR [Ref. 1.3.2].

Table 1.2-1 Regulatory Compliance Cross-Reference Matrix (3 Pages)

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-UMS CoC Amendment History	---	---
1.1.2 NAC-UMS 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	---	---

Table 1.2-1 Regulatory Compliance Cross-Reference Matrix (3 Pages)

CoC Renewal Application Section Number and Heading	NUREG-1927 Section Number and Heading	10CFR72 Requirement
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		
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 NAC-UMS 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	---	---
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		

Table 1.2-1 Regulatory Compliance Cross-Reference Matrix (3 Pages)

CoC Renewal Application Section Number and Heading	NUREG-1927 Section Number and Heading	10CFR72 Requirement
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 Operating Experience	---
3.6 Fuel Retrievability	---	§72.122(l)
3.7 Operating Experience Review	3.6.1.10 Operating Experience	---
3.8 Design Basis Document Review	---	---
3.9 References	---	---
Appendix A – Aging Management Program	Appendix A - Proposed Aging Management Programs	---
Appendix B – Time-Limited Aging Analysis	Appendix B – Time Limited Aging Analyses	§72.240(c)
Appendix C – NAC-UMS System FSAR Changes	1.4.4 Application Content	§72.240(c)
Appendix D – NAC-UMS System Technical Specification Changes	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-UMS Certificate No. 1015, Docket No. 72-1015:

a. NAC-UMS CoC; Initial Issue Amendment 0, Effective November 20, 2000.

b. NAC-UMS CoC; Amendment No. 1, Effective February 20, 2001.

c. NAC-UMS CoC; Amendment No. 2, Effective December 31, 2001.

d. NAC-UMS CoC; Amendment No. 3, Effective March 31, 2004.

e. NAC-UMS CoC; Amendment No. 4, Effective October 11, 2005.

f. NAC-UMS CoC; Amendment No. 5, Effective January 12, 2009.

g. NAC-UMS CoC; Amendment No. 6, Effective January 7, 2019.

h. NAC-UMS CoC; Amendment No. 7, Effective July 29, 2019.

1.3.2 NAC International, Inc., "Updated Final Safety Analysis Report for the NAC-UMS Universal Storage Cask System," Docket No. 72-1015, Revision 14, August 2019.

1.3.3 NAC International, Inc., Calculation No. 12412-9000, R0, "NAC-UMS Certificates of Compliance Reconciliation for Maine Yankee UMS 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 MY-COC-TSC-VCC-DFC for Maine Yankee Atomic Power Company, dated January 22, 2010.

1.3.5 Maine Yankee letter to USNRC MN-10-001, dated January 15, 2010, Request for Exemption from 10 CFR 72 (EGM-09-006).

1.3.6 USNRC letter to Maine Yankee, Exemption from 10 CFR 72.212 and 72.214 (TAC 24420), dated July 14, 2010.

1.3.7 Maine Yankee letter to USNRC OMY-11-131, MYAPC Adoption of NAC-UMS System, Amendment 5 Certificate of Compliance and Canister Registration, dated July 12, 2011.

1.3.8 NAC International, Inc., Calculation No. 12407-2005, R0, Arizona Public Service Palo Verde Nuclear Generating Station, "NAC-UMS Certificates of Compliance Reconciliation of Fabrication & Construction and Fuel Content Limitations for UMS-PWR Transportable Storage Canisters, Vertical Concrete Casks, and Transfer Casks," dated January 8, 2010.

1.3.9 NAC International, Inc. Supplemental Certificate of Conformance PV-COC-TSC/VCC 1-84 for Arizona Public Service – Palo Verde Nuclear Generating Station, dated January 14, 2010.

1.3.10 NAC International, Inc. Supplemental Certificate of Conformance PV-COC-TFR for NAC-UMS Advanced Transfer Cask, 790-060-95-1, Arizona Public Service – Palo Verde Nuclear Generating Station, dated January 14, 2010

1.3.11 Arizona Public Service letter to USNRC 102-06353-TNW/GAM, dated April 26, 2011, PVNGS Units 1, 2, and 3, and Independent Spent Fuel Storage Facility, Registration of Dry Spent Fuel Storage Casks with Applied Changes Authorized by an Amended Certificate of Compliance.

1.3.12 NAC Calculation No. 12419-2009, Duke Energy McGuire and Catawba Nuclear Stations, "NAC-UMS® Certificate of Compliance Amendment Reconciliation of Fabrication & Construction Activities, Operational Constraints and Fuel Contents Limitations UMS-PWR Transportable Storage Canisters, Vertical Concrete Casks & Transfer Casks", R0 dated July 3, 2007, and R1 dated January 12, 2010.

1.3.13 NAC International, Inc. Certificate of Conformance, Duke Energy – McGuire and Catawba Nuclear Stations, dated January 14, 2010.

1.3.14 NAC International, Inc. Supplemental Certificate of Conformance, Duke Energy – McGuire Nuclear Station, Duke Energy NAC-UMS Units 1-24, dated January 14, 2010.

1.3.15 NAC International, Inc. Supplemental Certificate of Conformance, Duke Energy – McGuire Nuclear Station, NAC-UMS Transfer Cask, Serial No. 01-1235-01, dated January 14, 2010.

1.3.16 NAC International, Inc. Supplemental Certificate of Conformance, Duke Energy – Catawba Nuclear Station, Duke Energy NAC-UMS Units 25-48, dated January 14, 2010.

1.3.17 NAC International, Inc. Supplemental Certificate of Conformance, Maine Yankee Atomic Power Company (Original Owner) and Duke Energy – Catawba Nuclear Station (Current Owner), NAC-UMS Transfer Cask, Unit MY-790-066-99, dated January 14, 2010.

1.3.18 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.

1.3.19 NEI-14-03, "Guidance for Operations Based Aging Management for Dry Cask Storage," Revision 2, December 2016.

1.3.20 Duke Energy Email, William J. Murphy to G. Tjersland, NAC, dated November 5, 2015.

1.3.21 Maine Yankee letter to USNRC OMY-19-020, MYAPC Adoption of NAC-UMS System, Amendment 6 Certificate of Compliance and Canister Registration, dated December 11, 2019

2.0 SCOPING EVALUATION

2.1 INTRODUCTION

The NAC-UMS System CoC renewal methodology follows NUREG-1927 [2.7.2] and NEI 14-03 [2.7.6]. The 10 CFR Part 72 license renewal process adopts the regulatory philosophy of 10 CFR Part 54. This philosophy is summarized in the two principles of license renewal from 10 CFR Part 54 Final Rule Statements of Consideration [2.7.3] which are re-stated below:

> *“The first principle of license renewal was that, with the exception of age-related degradation unique to license renewal and possibly a few other issues related to safety only during the period of extended 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 license renewal. …”*

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

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-UMS System will be carried forward through the renewed 40-year CoC renewal period.

The scoping process involves identification of the SSCs of the NAC-UMS 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 license renewal process involves the identification of the in-scope NAC-UMS System SSCs. This is done by evaluating the SSCs that comprise the NAC-UMS System against the following scoping criteria provided in NUREG-1927 [Reference 2.7.2].

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:
   (1) Subcriticality (CR),
   (2) Thermal/heat removal (TH),
   (3) Confinement (CO),
   (4) Radiation shielding (SH),
   (5) Structural integrity (SR), and
   (6) 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-UMS 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 [2.7.2] the NAC-UMS 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-UMS System.

- NAC-UMS System UFSAR [2.7.1.a thru o]
- CoC No. 1015 [2.7.4.a thru h]

The UFSAR provides a description of the cask system, SSCs and their functions, including safety classifications as established by the safety analysis. The applicable NAC-UMS System License Drawings utilized in the scoping process and contained in the approved FSARs are listed in Table 2.2-1. The CoC and associated Technical Specifications, govern the storage of irradiated nuclear fuel in the NAC-UMS System, and the transfer of such irradiated fuel to and from the spent fuel pool (SFP) and the ISFSI storage pad. Additionally, the Safety Evaluation Reports (SERs) [2.7.5. a thru h], which summarizes the results of the staff's safety review of the original licensing, and SERs associated with subsequent amendments were considered in the license renewal scoping process.

2.3 SCOPING RESULTS

The SSCs comprising the NAC-UMS 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 license renewal.

As indicated in Table 2.3-1, the Transportable Storage Canister (TSC), Vertical Storage Cask (VCC), Transfer Cask (TFR)/Transfer Adapter, and Spent Fuel Assemblies (SFA) were determined to be ITS and therefore, within the scope of license 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 do not meet Criterion 2 include Fuel Transfer Equipment, Ancillary Operating Systems, Temperature Monitoring Equipment, ISFSI Security Equipment, and other utility services or equipment.

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 could impact another SSC subcomponents' passive safety function 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-UMS System is a spent fuel dry storage system that uses a VCC and a stainless steel TSC with a double welded closure to safely store spent fuel. The NAC-UMS System is designed to provide long-term safe storage and subsequent transport of up to 24 PWR or up to 56 BWR spent fuel assemblies stored in the TSC in the NRC-certified NAC-UMS 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 in-scope SSCs of the NAC-UMS System are:

- TSC with PWR or BWR Fuel Basket (and Damaged Fuel Cans when authorized)
- VCC
- Transfer Cask/Transfer Adapter
- Spent Fuel Assemblies
- 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)
- Temperature Monitoring Equipment
- ISFSI Storage Pad
- ISFSI Security Equipment

License Drawings of the NAC-UMS System components and equipment are provided in the SAR and UFSAR that correspond with the initial CoC and all approved CoC amendments. A summary of the License Drawings included in each SAR and UFSAR revision associated with the initial CoC and all subsequent amendments is provided in Table 2.2-1. 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 TSC and integral fuel baskets are described in Sections 1.2.1.1 and 1.2.1.2 of the NAC-UMS System UFSARs [2.7.1.a thru o]. Three classes (e.g., length) of TSCs accommodate PWR SFAs, and two classes of TSCs accommodate the BWR SFAs.

The TSC consists of a stainless-steel canister that contains the fuel basket structure and contents. The canister is defined as the confinement pressure boundary for the spent fuel during storage and is provided with a double welded closure system. The leak tight TSC assembly ensures that the SNF contents are maintained in an inert helium atmosphere during handling, transfer and storage operations. The welded closure system of the TSC assembly prevents the release of contents in any design basis normal, off-normal or accident condition. The basket assembly in the canister provides the structural support and primary heat transfer path for the fuel assemblies while maintaining a subcritical configuration for all normal conditions of storage, off-normal events and hypothetical accident conditions. The major components of the TSC are the shell and bottom, basket assembly, shield lid, port covers, and structural lid. The canister and the shield and structural lids and port covers provide a confinement pressure boundary during storage, radiation shielding, and lifting capability for the basket.

The canister consists of a cylindrical Type 304L stainless steel shell with a Type 304L stainless steel bottom plate and a Type 304 stainless steel shield lid support ring. A basket assembly is placed inside the canister. The shield lid assembly is a Type 304 stainless steel disk that is positioned on the shield lid support ring above the basket assembly. The shield lid is welded to the canister after the canister is loaded and moved to the workstation for completion of canister closure activities. Two penetrations through the shield lid are provided for draining, vacuum drying, and backfilling the canister with helium. The drainpipe is threaded into the shield lid after the canister is moved to the workstation. The vent penetration in the shield lid is used to aid water removal and for vacuum drying and backfilling the canister with helium. After the shield lid is welded in place, it is pressure tested, and following port cover welding is helium leakage tested to leak-tight criteria to ensure no credible leakage of the confinement boundary during storage.

The structural lid is a Type 304L stainless steel disk positioned on top of the shield lid and welded to the shell after the shield lid is welded in place and the canister is drained, dried, and backfilled with helium. Removable lifting fixtures, installed in the structural lid, are used to lift and lower the loaded canister.

The TSC is designed to the requirements of the ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Division I, Subsection NB. It is fabricated and assembled in accordance with the requirements of Subsection NB consistent with NRC approved exemptions identified in the Technical Specifications.

Each TSC contains a fuel basket assembly which positions and supports the stored fuel in normal, off-normal and accident conditions. The design of the fuel basket is similar for the PWR and BWR configurations. The fuel basket assembly for each fuel type is designed and fabricated to the requirements of the ASME Code, Section III, Division I, Subsection NG consistent with NRC approved exemptions identified in the Technical Specifications. The fuel baskets are mainly constructed of stainless steel except for carbon steel support disks for the BWR fuel basket assemblies and aluminum disks for enhanced heat transfer. The fuel basket design is a right-circular cylinder configuration with square fuel tubes laterally supported by a series of support disks. The Class 1, 2 or 3 PWR fuel baskets incorporate 30, 32 or 34 support disks, respectively, and Class 4 or 5 BWR fuel baskets incorporate 40 or 41 support disks, respectively. The support disks are retained by a top nut and supported by spacers on tie rods at eight locations (PWR) or six locations (BWR). The top nut is torqued and welded at installation to provide a solid load path in compression between the support disks. The PWR support disks are fabricated of SA693, Type 630, 17-4 PH stainless steel and the BWR support disks are fabricated from SA533, Type B, Class 2 carbon steel. The PWR support disks are spaced axially at 4.92 inches center-to-center and BWR disks support are spaced at 3.7 inches. Each support disk contains either 24 (PWR) or 56 (BWR) square holes for the fuel tubes. The support disks and the stainless steel upper and bottom weldments, tie-rods and spacers, and top and bottom nuts provide the structural strength to maintain the basket assembly. The support disks and weldments also provide a criticality safety function by maintaining the spacing between adjacent assemblies and thereby maintaining a “flux trap” when the TSC and basket assembly is flooded with water.

The top and bottom weldments are fabricated from Type 304 stainless steel and are geometrically like the support disks. The tie rods and top nuts are fabricated from SA479, Type 304 stainless steel. The top nut is fabricated from a 3.5-in.-diameter bar, and the spacers are fabricated from a 2.5-in. Type 304 stainless steel pipe. The fuel tubes are fabricated from A240, Type 304 stainless steel and support an enclosed neutron absorber sheet on each of the four sides (PWR) or on two, one, or no sides (BWR). The neutron absorber provides criticality control in the basket. No credit is taken for the fuel tubes for structural strength of the basket or support of the fuel assemblies. As noted above the structural disks provide for the maintenance of the fuel basket configuration and the required flux trap spacing.

Each PWR fuel basket has a capacity of 24 PWR fuel assemblies in an aligned configuration in 8.80-inch square fuel tubes. The holes in the top weldment are 8.75-inch square. The holes in the bottom weldment are 8.65-inch square. The basket design traps the fuel tube between the top and bottom weldments, thereby preventing axial movement of the fuel tube. The PWR support disk configuration includes webs between the fuel tubes with variable widths depending on location. Each BWR fuel basket has a capacity of 56 BWR fuel assemblies in an aligned configuration. The fuel tubes in 52 positions have an inside square dimension of 5.90 inches. The inside dimension of the four fuel tubes located in the outside corners of the basket array is 6.05-inches square. The holes in the top weldment are 5.75 inches by 5.75 inches, except for the four enlarged holes, which are 5.90 inches square. The holes in the bottom weldment are 5.63-inches square. The basket design traps the fuel tube between the top and bottom weldments, thereby preventing axial movement of the fuel tube. The support disk webs between the fuel tubes are 0.65-inch wide.

The PWR and BWR basket design incorporates Type 6061-T651 aluminum alloy heat transfer disks to enhance heat transfer in the basket. Class 1, 2 and 3 PWR baskets contain 29, 31 or 33 disks respectively, which are spaced and retained between the structural disks. Class 4 and 5 BWR baskets both contain 17 disks, which are in the center of the basket assembly between structural disks where the need for heat transfer is greatest. The heat transfer disks are spaced and supported by the tie rods and spacers, which also support and locate the support disks. The heat transfer disks, located at the center of the axial spacing between the support disks, are sized to eliminate contact with the canister inner shell due to differential thermal expansion. No structural credit is taken for the aluminum alloy heat transfer disks.

The TSC and fuel baskets are designed to facilitate filling with water and subsequent draining. Water fills and drains freely between the basket disks through three separate paths.

One path is the gaps that exist between the disks and canister shell. The second path is through the gaps between the fuel tubes and disks that surrounds the fuel tubes. The third path is through three 1.3-inch diameter holes (PWR) or one 2.05-inch diameter hole (BWR) in each of the disks that provide additional paths for water flow between disks. The basket bottom weldment supports the fuel tubes above the canister bottom plate. The fuel tubes are open at the top and bottom ends, allowing the free flow of water from the bottom of the fuel tube. The bottom weldment is positioned by supports 1.0 inch above the canister bottom to facilitate water flow to the drain line. These design features ensure that water flows freely in the basket so that the canister fills and drains evenly. An optional recessed sump in the TSC baseplates of some of the deployed TSCs also facilitated removal of water from the canister.

2.4.3 Vertical Concrete Cask (VCC)

The VCC is the storage overpack for the TSC and is described in Section 1.2.1.3 of the NAC-UMS System UFSARs [2.7.1.a thru o]. Five VCCs of different lengths are designed to store the five TSCs of different lengths containing one of three classes of PWR or one of two classes of BWR SFAs. The VCC provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the canister during long-term storage. The VCC is a reinforced concrete (Type II Portland cement) structure with a structural steel inner liner. The concrete wall and steel liner provide the 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 VCC forms an annular air passage to allow the natural circulation of air around the TSC to remove the decay heat from the spent fuel. The air inlets and outlets are steel-lined penetrations that take nonplanar paths to the VCC cavity to minimize radiation streaming. A baffle assembly directs inlet air upward and around the pedestal that supports the canister. There is a stainless-steel sheet placed on top of the pedestal to prevent interaction of the stainless-steel baseplate of the TSC with the carbon steel pedestal. The weldment structure includes the baffle assembly configuration. The decay heat is transferred from the fuel assemblies to the tubes 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 of the 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 top of the VCC is closed by a shield plug and lid. The shield plug is approximately 5 inches thick and incorporates carbon steel plate as gamma radiation shielding, and NS-4-FR or NS-3 as neutron radiation shielding. A carbon steel lid that provides additional gamma radiation shielding is installed and bolted in place above the shield plug. The shield plug and lid reduce skyshine radiation and provide a cover and seal to protect the canister from the environment and postulated tornado missiles. At the option of the user, a tamper-indicating seal wire and seal may be installed on two of the concrete cask lid bolts. An optional supplemental shielding fixture may be installed in the air inlets to reduce the localized radiation dose rate at the air inlets of the VCC.

2.4.4 Transfer Cask (TFR) and Transfer Adapter

The transfer cask is a special lifting device, which is designed, fabricated, and load-tested to meet the requirements of NUREG-0612 [2.7.7] and ANSI N14.6 [2.7.8]. The description of the TFR is provided in Section 1.2.1.4 of the NAC-UMS UFSARs [2.7.1.a thru o]. The NAC-UMS System TFRs can be in a standard or advanced configuration, although all NAC-UMS System TFRs fabricated to date are advanced. The standard and advanced TFRs are designed for lifting and handling in the vertical orientation only. The advanced TFR is like the standard TFR, except that the advanced TFR incorporates a trunnion support plate that allows the advanced TFR to lift canisters weighing up to 98,000 pounds. The standard and advanced TFR each have four lifting trunnions, which allow for redundant load path lifting. Each set of two lifting trunnions is designed to provide for single-failure-proof handling of the TFR and TSC based on high design safety factors. Both TFR designs incorporate a multiwall (0.75-inch-thick inner carbon steel shell/4.0 inch lead brick gamma shielding/2.75 inch NS-4-FR solid neutron shield/1.25 inch thick outer carbon steel shell) design, and both designs have a maximum empty weight of approximately 121,500 pounds. The TFR overall height ranges from 187.8 to 204.5 inches depending on the class of TSC to be handled. The outer shell diameter is 85.25 inches with a trunnion to trunnion dimension of 95.3 inches. The inside diameter of the TFR is 67.25 inches (nominal) which provides a TSC to TFR annulus gap of approximately 0.35 inches. The TFR is coated with a spent fuel pool compatible coating system except for the door rails and interfacing door surfaces to minimize interaction with the spent fuel pool water and to facilitate decontamination. The shield doors are operated by hydraulic connectors located on the transfer adapter to allow for the vertical transfer of the TSC from or to the TFR and to or from the VCC or transport cask. The door rails and interfacing door surfaces are coated with spent fuel compatible grease to facilitate operation of the doors.

The TFR provides biological shielding when it contains a loaded canister and is used for the vertical handling and transfer of the canister between workstations and the concrete cask, or transport cask. Five transfer casks of either configuration, having different lengths, are designed to handle the five canisters of different lengths containing one of three classes of PWR fuel assemblies or one of two classes of BWR fuel assemblies. In addition, a Transfer Cask Extension may be used to extend the operational height. This height extension allows a transfer cask designed for a specific canister class to be used with the next longer canister.

The TFR design incorporates a top retaining ring, which is bolted in place to prevent a loaded canister from being inadvertently lifted through the top of the transfer cask. The transfer cask has retractable bottom shield doors. During loading operations, the doors are closed and secured by door lock pins, so they cannot inadvertently open. During unloading, 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, which also provides additional shielding for operational staff during TSC transfer operations.

To minimize the potential for contamination of a canister's externals or the inside of the TFR during loading operations in the spent fuel pool, clean filtered pool water is circulated in the annular gap between the TFR interior surfaces and the canister exterior surfaces. The TFR has eight supply and two discharge lines passing through its wall. Normally, two of the lines are connected to allow clean water under pressure to flow into and through the annular gap to minimize potential for the intrusion of pool water when the canister is being loaded. Lines not used for clean water supply or discharge are capped. The eight supply lines can also be used for the introduction of forced air at the bottom of the transfer cask to achieve cooling of the canister contents. This allows the canister to remain in the transfer cask for an extended period, if necessary, during canister closing operations.

2.4.5 Spent Fuel Assemblies (SFAs)

The NAC-UMS System is designed to accommodate up to twenty-four (24) intact, unconsolidated, zirconium alloy clad Pressurized Water Reactor (PWR) SNF assemblies or up to fifty-six (56) intact, unconsolidated, zirconium alloy clad Boiling Water Reactor (BWR) SNF assemblies in each TSC. A wide range of PWR and BWR SNF assembly types can be accommodated by the NAC-UMS System as identified on Tables 2.1.1-1 and 2.1.2-1, respectively, of Chapter 2 of the UFSARs [2.7.1.a thru o].

The total decay heat of the PWR fuel or BWR fuel contents shall not exceed 23.0 kW (maximum of 958.3 Watts for PWR and 410.7 Watts for BWR for standard SFAs; and a maximum of 1.05 kW for MY site-specific preferential fuel loading patterns with center fuel locations limited to 0.867 kW). The maximum initial enrichment shall not exceed 5.0 wt% 235U for PWR and 4.8 wt% 235U for BWR fuel assemblies. The maximum assembly average burnup for PWR SFAs is 60,000 MWd/MTU and for BWR SFAs is 45,000 MWd/MTU.

Flow mixers (thimble plugs), in-core instrument thimbles, burnable poison rods, control components or RCCAs, or solid stainless-steel rods may be placed in PWR guide tubes as long as the maximum fuel assembly weights are not exceeded and no credit for soluble boron is taken.

The NAC-UMS System is also certified to accommodate up to twenty-four (24) site-specific Maine Yankee (MY) SFAs which includes damaged, consolidated, and undamaged Combustion Engineering (CE) 14 x 14 fuel assemblies. MY damaged and consolidated SFAs, and SFAs having burnups ≥ 45,000 MWd/MTU were placed in Damaged Fuel Cans (DFCs) which are preferentially loaded in the four corner locations of the basket. The SFA conditions and configurations evaluated are described in Section 2.1.3.1 of the UFSARs [2.7.1.a thru o]. HBU fuel assemblies (burnups ≥ 45,000 MWd/MTU) have been loaded into the NAC-UMS System TSCs at PVNGS following approval of CoC No. 1015 Amendment 5

[2.7.4.f] without DFCs in a standard loading pattern in accordance with the CoC Technical Specifications.

2.4.6 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) are not included as part of the NAC-UMS System certified in NRC CoC for the NAC-UMS System and as such, are not described in detail in the NAC-UMS System UFSARs [2.7.1.a thru o]. 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-UMS System UFSAR [2.7.1.a thru o]. Some of the fuel transfer and auxiliary equipment is also depicted in the operational schematics shown in Figure 1.1-2 of the NAC-UMS System UFSAR.

2.4.7 VCC Temperature Monitoring System

The NAC-UMS System 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-UMS System CoC [2.7.4.a thru h], and as such, is not described in detail in the NAC-UMS UFSARs [2.7.1.a thru o]. Typically, a 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 VCC heat removal system is designed to maintain stored fuel cladding and NAC-UMS System SSCs 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-UMS System ISFSI storage pad is not part of the NAC-UMS System certified by the NRC in the NAC-UMS System CoC No. 1015 [2.7.4.a thru h], and as such, is not described in detail in the NAC-UMS System UFSARs [2.7.1.a thru o]. A typical ISFSI storage pad layout is shown in Figure 1.4-1 of the NAC-UMS System UFSAR. The ISFSI storage pad is a steel-reinforced concrete slab that supports free-standing NAC-UMS casks. As discussed in Section 1.4 of the NAC-UMS System UFSAR the ISFSI storage pad is analyzed to support the loads from the NAC-UMS System casks. Some NAC-UMS System users have identified the ISFSI storage pad as ITS (Category C) components and will perform aging management 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, and monitoring equipment) are not part of the NAC-UMS System approved by the NAC-UMS CoC [2.7.4.a thru h], and as such, are not described in the NAC-UMS System UFSARs [2.7.1.a thru o]. A typical ISFSI pad layout, which identifies some of the ISFSI security equipment (e.g., fencing), is shown in Figure 1.4-1 of the NAC-UMS System UFSAR. 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-UMS System casks from performing their intended functions.

2.5 SSC WITHIN SCOPE OF CoC RENEWAL APPLICATION

The SSCs determined to be within the scope of renewal are the TSC, VCC, and Transfer Cask (TFR)/Transfer Adapter. These basic components are the only SSCs important to safety (ITS) addressed in the CoC Nos. 1015 (2.7.4.a thru h) under 10 CFR 72, Subpart L. The TSC, VCC, TFR, and transfer adapter all satisfy Criterion 1 of the scoping evaluation. The SNF which are sealed within the TSC and supported by the fuel basket are also as identified as an SSC within the scope of the renewal in accordance with NUREG-1927 [2.7.2].

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, 2.5-2, 2.5-3, and 2.5-4. The important passive safety functions are defined by the following:

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

The applicable license drawings of the UFSAR 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 SSC subcomponents ITS. The Criterion 2 review identified additional SSC subcomponents that will require evaluation as in-scope for the CoC renewal evaluations. Tables 2.5-1, 2.5-2, 2.5-3, and 2.5-4 identify the intended functions for the NAC-UMS System SSC subcomponents that require an aging management review whether under either Criterion 1 or 2. The tables of the SSC subcomponents ITS also identify those subcomponents that do not support or impact the SSC intended passive safety function and are, therefore, not subject to an aging management review.

2.6 SSC NOT WITHIN SCOPE OF CoC RENEWAL APPLICATION

The SSC that are not in the scope of 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 Criteria 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-UMS System certified by the NRC in the NAC-UMS CoC No. 1015 [2.7.4.a thru h] and as such, are not described in detail in the NAC-UMS System UFSARs [2.7.1.a thru o]. The failure of the fuel transfer and auxiliary equipment would not prevent the TSC, VCC, or TFR/transfer adapter from fulfilling their intended safety functions. Therefore, the fuel transfer and auxiliary equipment do not meet the scoping Criteria 2 and are not within the scope of the certificate of compliance renewal. The fuel transfer and auxiliary equipment are addressed in site-specific reviews.

2.6.2 VCC Temperature Monitoring Equipment

The NAC-UMS System 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-UMS System CoC Nos. 1015 [2.7.4.a thru h], and as such, is not described in detail in the NAC-UMS UFSARs [2.7.1.a thru o]. Typically, a 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 NAC-UMS System SSCs 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 the scoping Criteria 2 and are not within the scope of the certificate of compliance renewal.

2.6.3 ISFSI Storage Pad

The NAC-UMS System ISFSI storage pad is not part of the NAC-UMS System certified by the NRC in the NAC-UMS CoC No. 1015 [2.7.4.a thru h] under 10 CFR Part 72, Subpart L. The ISFSI storage pad provides free-standing support of the NAC-UMS System Casks. The generic requirements for the ISFSI physical parameters are addressed in the USFARs [2.7.1.a thru o] in the evaluation of VCC accident drops and the beyond design basis tip-over accident. The UFSAR 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 classified as ITS Category C by the General Licensee. Although not within the scope of NAC-UMS System CoC renewal, the aging management, 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. [2.7.2]

Table 2.2-1 Applicable NAC-UMS 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)	FSAR R13/ 14(1)
790-501	UMS Canister/ Basket Assembly Table	2/1	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
790-559	Transfer Adapter Assy.	3/3	3/3	4/3	7/4	7/4	7/4	7/4	7/4	7/4	7/4	7/4	7/4	7/4	7/4
790-560	Transfer Cask Assy.	7/5	7/5	10/5	16/6	17/7	17/7	17/7	17/7	17/7	17/7	17/7	17/7	17/7	17/7
790-561	VCC Weldment Structure	5/3	5/3	8/3	10/4	12/4	12/4	13/4	14/4	14/4	15/4	15/4	15/4	15/4	15/4
790-562	Reinforcing Bar and Concrete Placement	5/4	5/4	10/5	13/7	14/7	16/7	16/7	17/7	17/7	18/7	18/7	18/7	19/7	19/7
790-563	VCC Lid	3/1	3/1	3/1	4/1	4/1	4/1	4/1	5/1	5/1	6/1	6/1	6/1	6/1	6/1
790-564	VCC Shield Plug	4/1	4/1	5/2	7/3	7/3	7/3	7/3	7/3	7/3	8/3	8/3	8/3	8/3	8/3
790-565	VCC Nameplate	1/1	1/1	2/1	4/1	4/1	4/1	4/1	4/1	4/1	5/1	5/1	5/1	5/1	5/1
790-570	56 BWR Fuel Basket (FB) (2) Assembly	3/2	3/2	3/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2

Table 2.2-1 Applicable NAC-UMS 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)	FSAR R13/ 14(1)
790-571	BWR FB Bottom Weldment	2/1	2/1	2/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1	3/1
790-572	BWR FB Top Weldment	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1	4/1
790-573	BWR FB Support Disk and Details	6/1	6/1	7/1	7/1	7/1	7/1	7/1	7/1	8/1	8/1	8/1	8/1	8/1	8/1
790-574	BWR FB Heat Transfer 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	3/1
790-575	BWR Fuel Tube	4/2	4/2	7/2	9/2	10/2	10/2	10/2	10/2	10/2	10/2	10/2	10/2	10/2	10/2
790-581	PWR Fuel Tube	5/2	5/2	5/2	8/2	9/2	9/2	9/2	9/2	9/2	9/2	9/2	9/2	9/2	9/2
790-582	Canister Shell Weldment	6/1	6/1	7/1	11/2	11/2	12/2	12/2	12/2	12/2	12/2	12/2	12/2	12/2	12/2
790-583	TSC Drain Tube Assembly	4/1	4/1	4/1	7/1	7/1	8/1	8/1	8/1	8/1	8/1	8/1	8/1	8/1	8/1
790-584	Canister Details	9/2	9/2	12/3	17/3	18/3	19/3	19/3	20/3	20/3	20/3	20/3	20/3	20/3	20/3
790-585	TSC Assembly	7/2	7/2	10/2	15/3	18/3	19/3	19/3	19/3	20/3	21/3	21/3	22/3	22/3	22/3

Table 2.2-1 Applicable NAC-UMS 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)	FSAR R13/ 14(1)
790-587	Canister Spacer Shim	-	-	-	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1	1/1
790-590	Loaded VCC	1/1	1/1	4/2	5/2	5/2	5/2	5/2	5/2	6/2	7/2	7/2	7/2	8/2	8/2
790-591	PWR FB Bottom Weldment	2/1	2/1	3/1	6/2	6/2	6/2	6/2	6/2	6/2	6/2	6/2	6/2	6/2	6/2
790-592	PWR FB Top Weldment	4/1	4/1	6/1	8/1	8/1	8/1	8/1	8/1	8/1	8/1	8/1	8/1	8/1	8/1
790-593	PWR FB Support Disk and Details	4/1	4/1	4/1	7/2	7/2	7/2	7/2	7/2	7/2	7/2	7/2	7/2	7/2	7/2
790-594	PWR Heat Transfer Disk	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1
790-595	24 PWR Fuel Basket Assembly	4/2	4/2	6/2	9/2	9/2	10/2	10/2	10/2	10/2	10/2	10/2	10/2	10/2	10/2
790-605	Over-Sized BWR Fuel Tube	5/2	5/2	8/2	10/2	11/2	11/2	11/2	11/2	11/2	11/2	11/2	11/2	11/2	11/2
790-613	VCC Inlet Supplemental Shield	-	-	0/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1	2/1

Table 2.2-1 Applicable NAC-UMS 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)	FSAR R13/ 14(1)
790-617	TFR Door Stop	-	-	1/2	2/2	3/2	3/2	3/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2
412-501	MY Spent Fuel Can Assembly	-	2/1	2/1	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2	4/2
412-502	MY Fuel Can Details	-	2/4	2/4	4/6	5/6	6/6	6/6	6/6	6/6	6/6	6/6	6/6	6/6	6/6

Note:

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

Table 2.3-1 Summary of Scoping Evaluation Results for NAC-UMS 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 (3)	Yes	NA	Yes
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.
(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.
(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-UMS Transportable Storage Canister (TSC) Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Shell	Items 1-5, 8	790-582	SR, CO, RE	A	X	---	Yes
Bottom	Item 6	790-582	SR, CO, RE	A	X	---	Yes
Location Lug	Item 7	790-582	---	C	---	---	No
Weather Resistant Paint (Alignment Mark) on TSC Shell	Dwg. Note 2	790-582	---	NQ	---	---	No
Drain Tube Nipple	Item 1	790-583	---	C	---	---	No
Drain Tube	Items 2-6	790-583	---	C	---	---	No
Seal	Item 7	790-583	---	C	---	---	No
Shield Lid	Item 1	790-584	SR, CO, SH	B	X	---	Yes
Nipple	Item 2	790-584	---	C	---	---	No
Seal	Item 3	790-584	---	C	---	---	No
Structural Lid	Item 4	790-584	SR, RE	A	X	---	Yes
Port Cover	Item 5	790-584	CO	B	X	---	Yes
Shield Lid Support Ring	Item 6	790-584	SR, SH	B	X	---	Yes
Spacer Ring	Item 7	790-584	SR	C	X-	---	Yes
Key	Item 8	790-584	---	C	---	---	No
Weather Resistant Paint (Alignment Mark) on Structural Lid	Dwg. Note 2	790-584	---	NQ	---	---	No
Shield Lid Plug	Item 22	790-585	---	NQ	---	---	No

Table 2.5-1 Intended Functions of NAC-UMS Transportable Storage Canister (TSC) Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Structural Lid Plug	Item 23	790-585	---	NQ	---	---	No
Dowel Pin	Item 24	790-585	---	NQ	---	---	No
BWR Drain Tube Sleeve	Item 4	790-570	---	C	---	---	No
BWR Basket Flat Washer	Item 23	790-570	SR	C	X	---	Yes
BWR Bottom Fuel Basket (FB) (3) Plate/Disk	Item 1	790-571	SR	A	X	---	Yes
BWR Bottom FB Weldment Pad	Item 2	790-571	SR	A	X	---	Yes
BWR Bottom FB Weldment Support Plate	Item 3	790-571	SR	A	X	---	Yes
BWR Top FB Plate/Disk	Item 1	790-572	SR	A	X	---	Yes
BWR Top FB Weldment Ring	Item 2	790-572	SR	A	X	---	Yes
BWR Top FB Weldment Support Plate	Items 3-5	790-572	SR	A	X	---	Yes
BWR Top FB Weldment Baffle	Item 6	790-572	SR	A	X	---	Yes
BWR FB Support Disk	Item 1	790-573	SR	A	X	---	Yes
Spacer	Item 3	790-573	SR	A	X	---	Yes
Top Nut	Item 4	790-573	SR	A	X	---	Yes
Tie Rods	Items 5-6	790-573	SR	A	X	---	Yes
Top Spacer	Item 7	790-573	SR	A	X	---	Yes
Split Spacer	Item 8	790-573	SR	A	X	---	Yes

Table 2.5-1 Intended Functions of NAC-UMS Transportable Storage Canister (TSC) Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
BWR FB Heat Transfer Disk	Item 1	790-574	TH	A	X	---	Yes
BWR Fuel Tube	Items 1-2	790-575	CR	A	X	---	Yes
Neutron Absorber	Items 3-4	790-575	CR	A	X	---	Yes
Cladding	Items 5-6	790-575	SR, CR	A	X	---	Yes
Tube Flange	Item 7	790-575	SR	A	X	---	Yes
BWR Oversize Fuel Tube	Items 1-2	790-605	CR	A	X	---	Yes
Neutron Absorber	Items 3-4	790-605	CR	A	X	---	Yes
Cladding	Items 5-6	790-605	SR, CR	A	X	---	Yes
Tube Flange	Item 7	790-605	SR	A	X	---	Yes
PWR Fuel Tube	Items 1-3	790-581	CR	A	X	---	Yes
Neutron Absorber	Items 4-6	790-581	CR	A	X	---	Yes
Cladding	Items 7-9	790-581	SR, CR	A	X	---	Yes
Tube Flange	Item 10	790-581	SR	A	X	---	Yes
Spacer Shim	Items 1-6	790-587	---	C	---	---	No
PWR Bottom Fuel Basket Plate/Disk	Item 1	790-591	SR	A	X	---	Yes
PWR Bottom FB Weldment Center Support Plate	Item 2	790-591	SR	A	X	---	Yes
PWR Bottom FB Weldment Support Plate	Items 3, 5-7	790-591	SR	A	X	---	Yes
PWR Bottom FB Weldment Pad	Item 4	790-591	SR	A	X	---	Yes

Table 2.5-1 Intended Functions of NAC-UMS Transportable Storage Canister (TSC) Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
PWR Top FB Plate/Disk	Item 1	790-592	SR	A	X	---	Yes
PWR Top FB Weldment Ring	Item 2	790-592	SR	A	X	---	Yes
PWR Top FB Weldment Support Plate	Items 3, 5-6	790-592	SR	A	X	---	Yes
PWR Top FB Weldment Center Support Plate	Item 4	790-592	SR	A	X	---	Yes
PWR Top FB Weldment Baffle	Item 7	790-592	SR	A	X	---	Yes
PWR FB Support Disk	Item 1	790-593	SR	A	X	---	Yes
Split Spacer	Item 2	790-593	SR	A	X	---	Yes
Spacer	Item 3	790-593	SR	A	X	---	Yes
Top Nut	Item 4, 9-10	790-593	SR	A	X	---	Yes
Tie Rods	Items 5-7	790-593	SR	A	X	---	Yes
Top Spacer	Item 8	790-593	SR	A	X	---	Yes
PWR FB Heat Transfer Disk	Item 1	790-594	TH	A	X	---	Yes
PWR Drain Tube Sleeve	Item 4	790-595	---	C	---	---	No
PWR Basket Flat Washer	Item 8	790-595	SR	C	X	---	Yes
DFC Collar	Item 1	412-502	CO	A	X	---	Yes
DFC Lid Plate	Item 2	412-502	SR, CR	A	X	---	Yes
Lid Guide	Item 3	412-502	---	C	---	---	No
Wiper	Item 4	412-502	CO	C	X	---	Yes
Lid Bottom	Item 5	412-502	SR, CR	A	X	---	Yes

Table 2.5-1 Intended Functions of NAC-UMS Transportable Storage Canister (TSC) Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Function(s) (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Filter Screen	Items 6 & 14	412-502	CO	C	X	---	Yes
Backing Screen	Items 7 & 15	412-502	CO	C	X	---	Yes
Bottom Plate	Item 8 & 17	412-502	SR, CR	A	X	---	Yes
Side Plate	Item 9 & 18	412-502	SR, CR	A	X	---	Yes
DFC Tube Body	Items 10 & 19	412-502	SR, CR	A	X	---	Yes
Lift Tee	Item 12	412-502	SR	B	X	---	Yes
Support Ring	Item 13	412-502	SR	B	X	---	Yes
Dowel Pin	Item 16	412-502	SR	C	X	---	Yes

Notes:

(1) Included in Section 1.8 of the NAC-UMS System Updated Final Safety Analysis Report (UFSAR) [2.7.1.a thru o]

(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Sub-Criticality (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.

Table 2.5-2 Intended Functions of NAC-UMS Vertical Concrete Cask (VCC) Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
VCC Liner Shell	Items 1, 27-30	790-561	SH, TH, SR	B	X	---	Yes
Top Flange	Item 2	790-561	SR	B	X	---	Yes
Support Ring	Item 3	790-561	SR	C	X	---	Yes
Jack Base	Item 4	790-561	---	NQ	---	---	No
Jack Gusset	Item 5	790-561	---	NQ	---	---	No
Jack Screw	Item 6	790-561	---	NQ	---	---	No
Jack Nut	Item 7	790-561	---	NQ	---	---	No
Jack Jam Nut	Item 8	790-561	---	NQ	---	---	No
Base Weldment Inlet Cover	Item 10	790-561	SR, TH, SH	B	X	---	Yes
Base Weldment Shield Ring	Item 11	790-561	SR, TH, SH	B	X	---	Yes
Base Weldment Bottom	Item 12	790-561	SR, TH, SH	B	X	---	Yes
Inlet Side	Item 13	790-561	SR, TH, SH	B	X	---	Yes
Inlet Top	Item 14	790-561	SR, TH, SH	B	X	---	Yes
Stand Plate	Item 15	790-561	SR, TH, SH	B	X	---	Yes
Baffle Weldment Base Plate	Item 16	790-561	SR	B	X	---	Yes
Nelson Stud	Item 17	790-561	SR	B	X	---	Yes
Outlet Bottom	Items 18 & 21	790-561	SR, TH, SH	B	X	---	Yes
Outlet Top	Item 19 & 22	790-561	SR, TH, SH	B	X	---	Yes
Outlet Shield Plate	Item 20	790-561	SR, TH, SH	B	X	---	Yes
Outlet Side	Item 23	790-561	SR, TH, SH	B	X	---	Yes
Outlet Back	Item 24	790-561	SR, TH, SH	B	X	---	Yes
Baffle	Item 25	790-561	SR, TH, SH	B	X	---	Yes
Screen Tab	Item 26	790-561	SR, TH, SH	C	X	---	Yes

Table 2.5-2 Intended Functions of NAC-UMS Vertical Concrete Cask (VCC) Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Lifting Nut	Item 31	790-561	---	NQ	---	---	No
Supplemental Shield Pipe/Tube/Bar (4)	Item 35	790-561	SH	B	X	---	Yes
Alternate Baffle Weldment Cover (5)	Item 36	790-561	SR	C	X	---	Yes
Dowel Pin	Item 37	790-561	---	NQ	---	---	No
Primer and Coating for Liner, Pedestal and Baseplate Assemblies	Item 32 and Dwg. Note 3	790-561	---	NQ	---	---	No
Rebar / Threaded Rebar	Items 1-11, 33, 43, & 46	790-562	SR, SH	B	X	---	Yes
Concrete Shell	Item 15	790-562	SR, SH	B	X	---	Yes
Vent Screen	Item 16	790-562	SR, TH, SH	NQ	---	X	Yes
Screen Strips	Item 17	790-562	SR, TH, SH	NQ	---	X	Yes
Flat Washer	Items 18 & 38	790-562	---	NQ	---	---	No
Screen Screw	Items 19 & 37	790-562	---	NQ	---	---	No
Concrete Anchor	Items 28, 36 & 39	790-562	---	NQ	---	---	No
Screw	Item 29	790-562	---	NQ	---	---	No
Name Plate	Item 30	790-562	---	NQ	---	---	No
Lifting Lug	Item 31	790-562	SR	B	X	---	Yes
Anchor Base Plate	Item 32	790-562	SR	B	X	---	Yes
Screen Bolt	Item 40	790-562	---	NQ	---	---	No
Washer	Item 41	790-562	---	NQ	---	---	No
Supplemental Cover	Item 42	790-562	SR, TH, SH	NQ	---	X	Yes

Table 2.5-2 Intended Functions of NAC-UMS Vertical Concrete Cask (VCC) Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Nut	Item 44	790-562	SR	B	X	---	Yes
Washer	Item 45	790-562	SR	B	X	---	Yes
Spacer Plate	Item 47	790-562	SR	B	X	---	Yes
Retainer Plate	Item 48	790-562	SR, TH, SH	NQ	---	X	Yes
Screen	Item 49	790-562	SR, TH, SH	NQ		X	Yes
Alternate Screen	Item 50	790-562	SR, TH, SH	NQ		X	Yes
Primer and Coating for Lift Lugs/Anchors	Dwg. Note 19	790-562	---	NQ	---	---	No
VCC Lid	Item 1	790-563	SR	B	X	---	Yes
Primer and Paint for VCC Lid	Dwg. Note 1	790-563	---	NQ	---	---	No
Shield Plug Plate	Item 1	790-564	SR, SH	B	X	---	Yes
Neutron Shield Retaining Ring	Items 2, 6, 7	790-564	SR	B	X	---	Yes
Neutron Shield	Items 3 & 5	790-564	SH	B	X	---	Yes
Neutron Shield Cover Plate	Items 4 & 8	790-564	SR, SH	B	X	---	Yes
Lifting Boss	Item 9	790-564	SR	C	X	---	Yes
Center Boss	Item 10	790-564	SR	C	X	---	Yes
Primer and Coating for Shield Plug	Items 11-13 and Dwg. Notes 1 and 5	790-564	---	NQ	---	---	No
Lid Bolt	Item 13	790-590	SR	B	X	---	Yes
Washer	Item 14	790-590	---	NQ	---	---	No
Cover	Item 15	790-590	SR	C	X	---	Yes
Seal Tape	Item 16	790-590	---	NQ	---	---	No

Table 2.5-2 Intended Functions of NAC-UMS Vertical Concrete Cask (VCC) Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Security Seal (6)	Item 17	790-590	---	C	---	---	No
Seal Wire (6)	Item 18	790-590	---	C	---	---	No
Tab	Item 19	790-590	---	NQ	---	---	No
Removable Supplemental Shield Side Plate (4)	Item 1	790-613	SH	B	X	---	Yes
Supplemental Shield Pipe/Tube/Bar (4)	Item 2	790-613	SH	B	X	---	Yes
Coating	Item 3	790-613	---	NQ	---	---	No
Shims	Item 4	790-613	---	NQ	---	---	No

Notes:

(1) Included in Section 1.8 of the NAC-UMS System UFSAR [2.7.1.a thru o]
(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Sub-Criticality (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) If installed. Not all deployed systems have inlet vent supplemental shield assemblies.
(5) Alternate cover assembly. Aging management evaluated same as for standard cover assembly under Dwg. No. 790-590, Item 15.
(6) Items 17 and 18 are optional to install and therefore are not identified as in-scope.

Table 2.5-3 Intended Functions of NAC-UMS Transfer Cask (TFR) / Transfer Adapter Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Bottom Plate	Item 1	790-560	SR	B	X	---	Yes
TFR Inner Shell	Items 2-6	790-560	SR	B	X	---	Yes
TFR Outer Shell	Items 7-11	790-560	SR	B	X	---	Yes
Trunnion	Item 12	790-560	SR	B	X	---	Yes
Trunnion Cap	Item 13	790-560	---	C	---	---	No
Neutron Shield	Item 14	790-560	SH	B	X	---	Yes
Top Plate	Item 15	790-560	SR	B	X	---	Yes
Shield Door Rail	Item 16	790-560	SR, SH	B	X	---	Yes
Door Lock Bolt	Item 19	790-560	SR	C	X	---	Yes
Retaining Ring	Item 20	790-560	SR	B	X	---	Yes
Support Plate	Item 21	790-560	SR	B	X	---	Yes
Spent Fuel Pool Coating System	Item 22 and Dwg. Note 7	790-560	---	NQ	---	---	No
Lead Wool	23	790-560	---	NQ	---	---	No
Paint	Item 24	790-560	---	NQ	---	---	No
Nameplate	Item 25	790-560	---	NQ	---	---	No
Shield Door (SD) Bottom Plate A	Item 26	790-560	SR, SH	B	X	---	Yes
SD Bottom Plate B	Item 27	790-560	SR, SH	B	X	---	Yes
SD Neutron Shield Boundary Plate	Items 28-32	790-560	SR, SH	B	X	---	Yes
SD Neutron Shield Cover Plate A	Item 33	790-560	SR, SH	B	X	---	Yes
SD Neutron Shield Cover Plate B	Item 34	790-560	SR, SH	B	X	---	Yes

Table 2.5-3 Intended Functions of NAC-UMS Transfer Cask (TFR) / Transfer Adapter Subcomponents

Subcomponent	Part or I.D. No.	Reference Drawing (1)	Intended Safety Functions (2)	Quality Classification	Sub-Scoping Results		In-Scope (3)
					Criterion 1	Criterion 2	
Gamma Shield Brick	Item 36	790-560	SH	B	X	---	Yes
Scuff Plate	Item 37	790-560	---	NQ	---	---	No
Retaining Ring Bolt	Item 38	790-560	SR	B	X	---	Yes
Connector	Item 39	790-560	SR	C	X	---	Yes
TFR Extension	Item 41	790-560	SR, SH	B	X	---	Yes
TFR Extension Bolts	Item 42	790-560	SR	B	X	---	Yes
Shielding Ring	Item 43	790-560	SH	B	X	---	Yes
Fill/Drain Line Plate	Item 44	790-560	---	C	---	---	No
Fill/Drain Line Pipe	Item 45	790-560	---	C	---	---	No
Dowel Pin	Item 46	790-560	---	NQ	---	---	No
Door Lock Bolt	Item 47	790-560	SR	C	X	---	Yes
Wear Strip	Item 49	790-560	---	NQ	---	X	Yes
Door Plug	Item 50	790-560	---	NQ	---	---	No
Lift Plate A	Item 51	790-560	---	NQ	---	---	No
Lift Plate B	Item 52	790-560	---	NQ	---	---	No
TFR Door Stop Bottom Plate	Item 1	790-617	---	NQ	---	---	No
Top Plate	Item 2	790-617	---	NQ	---	---	No
Back Plate	Item 3	790-617	---	NQ	---	---	No
Handle	Item 4	790-617	---	NQ	---	---	No
Lock Pin	Item 5	790-617	SR	NQ	---	X	Yes
Attachment Screw	Item 6	790-617	---	NQ	---	---	No
Transfer Adapter	Items 1 – 10, 12, 15-18	790-559	SH, SR	C	X	---	Yes

Notes:

(1) Included in Section 1.8 of the NAC-UMS UFSAR

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

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

Table 2.5-4 Intended Functions of Spent Fuel Assembly[1] (SFA) Subcomponents in NAC-UMS Systems

Subcomponent	Part or I.D. No.	Reference Drawing [1]	Intended Safety Functions [2]	Quality Classification	Sub-Scoping Results		In-Scope
					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, CR	A	X	---	Yes
Spacer grids	NA	NA	CR, RE, SR, TH	A	X	---	Yes
Lower and upper end fittings	NA	NA	RE, SR, CR	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-UMS Systems described in Sections 1.3.1 of the NAC-UMS UFSAR [2.7.1.a thru o]
(2) Intended safety functions include Thermal/Heat Removal (TH), Structural Integrity (SR), Confinement (CO), Radiation Shielding (SH), Sub-Criticality (CR), and Retrievability (RE)

2.7 REFERENCES

2.7.1 NAC International, Inc., "Updated Final Safety Analysis Report for the NAC-UMS Universal Storage Cask System," Docket No. 72-1015:

a. NAC-UMS System Final Safety Analysis Report, Revision 0, December 2000.

b. NAC-UMS System Updated Final Safety Analysis Report, Revision 1, May 2001.

c. NAC-UMS System Updated Final Safety Analysis Report, Revision 2, January 2002.

d. NAC-UMS System Updated Final Safety Analysis Report, Revision 3, March 2004.

e. NAC-UMS System Updated Final Safety Analysis Report, Revision 4, November 2004.

f. NAC-UMS System Updated Final Safety Analysis Report, Revision 5, October 2005.

g. NAC-UMS System Updated Final Safety Analysis Report, Revision 6, November 2006.

h. NAC-UMS System Updated Final Safety Analysis Report, Revision 7, November 2008.

i. NAC-UMS System Updated Final Safety Analysis Report, Revision 8, February 2009.

j. NAC-UMS System Updated Final Safety Analysis Report, Revision 9, November 2010.

k. NAC-UMS System Updated Final Safety Analysis Report, Revision 10, October 2012.

l. NAC-UMS System Updated Final Safety Analysis Report, Revision 11, November 2016.

m. NAC-UMS System Updated Final Safety Analysis Report, Revision 12, November 2018.

n. NAC-UMS System Updated Final Safety Analysis Report, Revision 13, January 2019.

o. NAC-UMS System Updated Final Safety Analysis Report, Revision 14, July 2019.

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

2.7.3 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

2.7.4 U.S. Nuclear Regulatory Commission, Certificate of Compliance for Spent Fuel Storage Casks, Model No.: NAC-UMS Certificate No. 1015, Docket No. 72-1015:

a. NAC-UMS CoC; Initial Issue Amendment 0, Effective November 20, 2000.

b. NAC-UMS CoC; Amendment No. 1, Effective February 20, 2001.

c. NAC-UMS CoC; Amendment No. 2, Effective December 31, 2001.

d. NAC-UMS CoC; Amendment No. 3, Effective March 31, 2004.

e. NAC-UMS CoC; Amendment No. 4, Effective October 11, 2005.

f. NAC-UMS CoC; Amendment No. 5, Effective January 12, 2009.

g. NAC-UMS CoC; Amendment No. 6, Effective January 7, 2019.

h. NAC-UMS CoC; Amendment No. 7, Effective July 29, 2019.

2.7.5 Safety Evaluation Report (SER) for NAC-UMS System, Docket No. 72-1015, SER issued with CoC No. 1015

a. Safety Evaluation Report (SER) with an effective date of November 20, 2000.

b. Safety Evaluation Report (SER) with an effective date of February 20, 2001.

c. Safety Evaluation Report (SER) with an effective date of December 31, 2001.

d. Safety Evaluation Report (SER) with an effective date of March 31, 2004.

e. Safety Evaluation Report (SER) with an effective date of October 11, 2005.

f. Safety Evaluation Report (SER) with an effective date of January 12, 2009.

g. Safety Evaluation Report (SER) with an effective date of January 7, 2019.

h. Safety Evaluation Report (SER) with an effective date of July 29, 2019.

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

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

2.7.8 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-UMS 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-UMS System SSCs within the scope of license renewal which require evaluation for the effects of aging in the aging management review process. The methodology used for the AMR of the NAC-UMS 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-UMS 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-UMS System operating experience and pre-application inspection report are presented in Section 3.7 and a discussion of the design basis document review efforts is presented in Section 3.8. The results of the AMR are summarized in Table 3.2-1 through 3.2-4. 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-UMS System SSCs that require aging management review (AMR), although they do not identify potential aging effects or mechanisms, or specific aging management methods. 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-UMS System UFSARs [3.9.1.a thru o] 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-UMS System UFSAR [3.9.1], 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 3.2-4. 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 NAC-UMS System 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-UMS System and 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 major components of the 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. 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 TSC shell is fabricated from a cylindrically rolled, 5/8-inch-thick (0.625 in.) SA240, Type 304L stainless steel plate. The nominal external diameter of the TSC shell is 67.06 inches with a 65.81-inch nominal internal diameter. The shell is formed with a full penetration weld. If the TSC shell requires a girth weld, the seam welds of adjacent shell sections are offset approximately 45º. The TSC shell seam and girth welds are 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 is welded to a SA240, Type 304L stainless steel, 1 3/4-inch-thick base plate with a full penetration weld. The NDE of the TSC shell to the base plate weld is 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. To secure the basket assembly axially in the TSC shell assembly, and to position the TSC shield lid for welding, a SA479/SA240, Type 304 stainless steel, 1/2 x 1/2-inch-square lid support ring is installed, positioned and welded to the TSC shell above the basket assembly top weldment. Additionally, an ASTM A240/A276, Type 304 stainless steel 4 1/2-inch-long x 1-inch-wide x 1/2-inch-high key is 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.

For each TSC shell assembly, a unique TSC shield lid, structural lid, port covers, and drain tube assembly are fabricated. The TSC shield lid is a SA240/SA182, Type 304 stainless steel, 7-inch-thick, 65.5-inch-diameter plate/forging that is installed on a loaded TSC assembly underwater, rests on the lid support ring, and is rotationally aligned by the key. Following removal of the TFR from the pool, the TSC shield lid is welded to the TSC shell with a 3/8-inch-thick (1/2-inch for BWR fuel TSCs), multi-pass partial penetration weld. NDE of the TSC shield lid-to-TSC shell weld is 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 may be used to reduce the weld gap during shield lid-to-TSC shell welding operation.

The TSC shield lid is 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. The drain and vent valved nipples are sealed to the TSC shield lid threaded openings using stainless steel, Viton or EDPM polymer seals. No confinement credit is taken by the quick-disconnect valved nipples and seals during storage operations.

Following pressure testing, draining, drying and backfilling of the cavity with helium, the vent and drain openings are closed by welding in place SA479, Type 304 stainless steel, 3.4-inch high x 5.9-inch diameter port covers that fit around the valved nipple and fill the penetration volume to minimize radiation streaming. The port covers are 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 is tested for helium leakage to leaktight criteria in accordance with ANSI N14.5 [3.9.26] requirements. The TSC shield lid is provided with three, 1-8 UNC-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 TSC structural lid is installed on top of the shield lid. The TSC structural lid is a SA240/SA182, Type 304L 3-inch-thick, 65 1/2-inch- diameter stainless steel plate/forging. An SA479/SA240, Type 304 1/2 x 1/2-inch stainless steel spacer ring is 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 3/4-inch (7/8-inch for BWR TSCs) multi-pass partial penetration weld performed with progressive VT and PT examinations of the root, each intermediate weld layer (not exceeding 3/8-inch), and the final weld surface. The PT examinations are 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 1/2 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.

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 and neutron absorbing poison materials.

Each PWR fuel basket is an assembled structure of stainless-steel disks and weldments installed on eight tie rods. Aluminum heat rejection disks are interspersed with the stainless-steel support disks in an alternating pattern. The PWR fuel basket assembly is a right-circular configuration with twenty-four (24) square fuel tubes laterally supported by the support disks and weldments, and axially restrained by the top and bottom weldments. The basket is assembled on eight 1 5/8-inch-diameter tie-rods fabricated from SA479, Type 304 stainless steel bar. The 1-inch-thick bottom weldment, fabricated from SA240, Type 304 stainless steel, is installed on the eight tie rods and is positioned axially by eight, SA479/SA240, Type 304 stainless steel, 1-inch-thick, 3 1/2-inch-diameter support pads that are welded to the base of the bottom weldment. Additionally, SA240/SA479, Type 304 stainless steel 1-inch-thick by 1-inch-high supports are welded to the base of the bottom weldment to axially position the PWR basket assembly off the bottom of the TSC to facilitate the draining of cavity water.

The basket assembly operation is continued with the installation of the first 1/2-inch-thick SA693, Type 630, 17-4 PH stainless steel support disk on the eight tie rods. The support disk is positioned 4.42 inches off the bottom weldment by the use of the 4.42-inch high spacers fabricated from SA312, Type 304 stainless steel that fit around each of the tie rods. Each support disk has twenty-four 9.27-inch-square holes machined to accommodate the fuel tubes. The assembly is continued with the installation of 2.2-inch-high SA312, Type 304 stainless steel split spacers combined with 2 1/2-inch-diameter stainless steel flat washers that position the 1/2-inch-thick Type 6061-T651 aluminum alloy heat transfer disks between each support disk. Each heat transfer disk has twenty-four 9.24-inch-square holes machined to accommodate the fuel tubes. The basket assembly is continued by alternating support disks and heat transfer disks using the split spacers and flat washers. After installation of the top-most support disk, the 24 fuel tubes are 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 (8.65-inch-square) machined openings. Each fuel tube has four sheets of neutron absorber held in place on the exterior of the tube by stainless steel sheathing (A240, Type 304). Each sheet of neutron absorber has a minimum B10 areal density of 0.025g/cm^2. The eight 4.42-inch high top spacers are then used to position the 1 1/4-inch-thick SA240, Type 304 stainless steel top weldment 4.42 inches from the top support disk. The top weldment is reinforced by a 1/2-inch-thick SA240, Type 304 stainless steel ring and 1-inch-thick SA240, Type 304 stainless steel support plates. The top weldment is held in place by eight SA479, Type 304 stainless steel top nuts, fabricated from 3 1/2-inch bar, that are installed on the eight tie rods. The height of the top nuts varies for the class of PWR fuel basket and height of the top weldment. Following torquing, the top nuts are welded to the top weldment to prevent loosening.

Like the PWR fuel basket, the BWR fuel basket is an assembled structure of stainless-steel top and bottom weldments and carbon steel support disks installed on six tie rods. Aluminum heat transfer disks are interspersed in the central region of the basket assembly with the carbon steel support disks in an alternating pattern. The heat transfer disks provide improved heat transfer capabilities in the high heat load central section of the BWR fuel basket assembly. The BWR fuel basket assembly is a right-circular configuration with fifty-six (56) square fuel tubes laterally supported by the support disks and weldments, and axially restrained by the top and bottom weldments.

Like the PWR basket, the BWR basket is assembled on six 1 5/8-inch-diameter tie-rods fabricated from SA479, Type 304 stainless steel bar. The 1-inch-thick bottom weldment, fabricated from SA240, Type 304 stainless steel, is installed on the six tie rods and retained in axial position by welding to six SA479, Type 304 stainless steel, 4-inch-high and 3-inch in diameter support pads that are welded to the base of the bottom weldment. Additionally, six SA240SA479, Type 304 stainless steel 3/4-inch-thick by 4-inch-high by 25-inch support plates are welded to the base of the bottom weldment. The basket assembly operation is continued with the installation of the first 5/8-inch-thick SA533, Type B, Class 2 electroless nickel-plated carbon steel support disk on the six tie rods. The support disk is positioned 3.2 inches off the bottom weldment and between adjacent support disks outside of the region provided with heat transfer disks using six 3.2-inch high spacers fabricated from SA312, Type 304 stainless steel bar with a 1.8-inch drilled hole that fit around the tie rods. The use of two 1.6-inch high split spacers in locations provided with heat transfer disks resulting in a center-to-center spacing of 3.8 inches for the support disks. Each support disk has 52 6.28-inch-square and four 6.43-inch-square machined holes to accommodate 52 standard BWR fuel tubes having 5.90-inch-square openings and four fuel tubes having a larger 6.05-inch-square opening. In the center region of the BWR fuel basket, Type 6061-T6 aluminum alloy heat transfer disks are installed between the support disks. Each heat transfer disk has 52 6.28-inch-square openings machined for the standard BWR fuel tubes and four 6.428 inch-square openings machined for the four larger cross section BWR fuel tubes.

The BWR basket assembly is continued by alternating support disks and heat transfer disks using 2.88-inch-diameter by 1.6-inch-high SA312, Type 304 stainless steel split spacers with a 1.8-inch drilled hole. The split spacers are installed on each of the six tie rods, followed by 2 1/2-inch-diameter stainless steel flat washers. The aluminum heat transfer disk is then installed followed by an additional flat washer, split spacer and support disk. The above-described process is then continued until the 17 heat transfer disks are installed. The installation of the 3.2-inch spacers alternating with support disks is then continued until all the remaining support disks are installed. The 52 standard and 4 oversized A240, Type 304 stainless steel BWR fuel tubes are then installed in their appropriate location following the neutron absorber sheet pattern as presented on the basket assembly drawing. The neutron absorber is retained in place on the outer surface of the tube by A240, Type 304 stainless steel sheathing, which is welded to the stainless-steel tube surface. The BWR fuel tubes have neutron absorber on two, one or no sides of the fuel tube dependent on its location in the BWR fuel basket assembly.

The 1-inch-thick SA240, Type 304 stainless steel top weldment is then installed on the tie rods above the final spacers. The top weldment is reinforced by a 3/8-inch-thick SA240/SA479, Type 304 stainless steel ring and by eight 1/2-inch-thick SA240/SA479, Type 304 stainless steel support plates. The top weldment is held in place by six 10.3-inch-high SA479, Type 304 stainless steel top nuts fabricated from 3-inch-diameter bar. Following torquing of the top nuts, the top nuts are welded to the top weldment to prevent loosening.

3.1.1.2 Vertical Concrete Cask (VCC)

The NAC-UMS System VCC is the storage overpack for the TSC and 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 #6 ASTM A615/A615M carbon steel rebar. The internal cavity of the VCC assembly is lined by the 2-1/2-inch-thick ASTM A36 carbon steel liner with a 2-inch-thick top flange and 2-1/2 x 3-inch shield ring. 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 which incorporates a 2-inch thick ASTM A36 carbon steel base plate which supports the stored TSC. 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 fabricated from 3-3/4 inch and 3/8-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 1-1/2-inch-thick ASTM A36 carbon steel VCC lid bolted to the top lid by six stainless steel hex head bolts. Two lifting lugs fabricated from 2-inch-thick ASTM A537, Class 2 carbon steel can be provided at the option of the user and are embedded into the top of the concrete surface. Exposed surfaces of the VCC carbon steel not covered by the concrete shell are 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 top coat, or equivalent. The VCC assembly also includes a Type 304 stainless steel sheet on the top of the baffle base plate to support the loaded stainless steel TSC and to prevent contact with the carbon steel baffle base plate surface. 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 Grade B or A106 Grade B for 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 / Transfer Adapter

The NAC-UMS System Transfer Casks (TFR) are a special lifting device designed, fabricated, tested, and maintained to meet the requirements of NUREG-0612 [3.9.24] and ANSI N14.6 [3.9.25]. 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, and trunnion cap). The 3/4-inch-thick inner radial shell, 1-1/4 inch outer radial shell, and 3.2 inch thick top and 1 inch thick bottom plates form an annulus into which the approximately 4.0 inch thick lead gamma shield bricks (ASTM A20) are assembled and interlocked. NS-4-FR neutron shielding material is then poured in place to form a 2.7-inch-thick layer before final closure of the cavity. Additional TFR components are constructed of ASTM A350 LF2 low alloy steel (e.g., shield door assemblies' bottom plates [total shield door thickness of 9 inches including 1.5-inch-thick layer of NS-4-FR neutron shielding], door rails, and lifting trunnions). The door rails are welded to the lower plate of the main body and support the two shield doors. The four 10-inch diameter lifting trunnions penetrate through the inner and outer shells near the top of the cask body and are welded to the inner and outer shells. For the advanced TFR design, an ASTM A36/A105/A516 Grade 70 shield ring is installed above the lifting trunnions prior to closure of the inner and outer shells. The TFR also features an ASTM A588 low alloy steel 3/4-inch-thick retaining ring bolted to the upper plate by ASTM A193, Grade B6 bolts which prevents the TSC from being accidently removed from the TFR annulus during the loaded TSC transfer operation. In addition, a Class 1/2 extension constructed of ASTM A516, Grade 70 carbon steel is available for use at sites using both Class 1 and Class 2 TSCs without the need for two separate TFRs. The extension is retained to the cask body by ASTM A193, Grade B6 socket head cap screws, and accommodates the fit-up of the retaining ring when used for TSC transfer operations. 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 TFR and its 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 TFR may be provided with Nitronic 30 wear strips on the inner shell to minimize coating repairs. The coated surfaces around the trunnions are protected by stainless steel scuff plates 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 lubricated with a spent fuel compatible lubricant such as Neolube or equivalent to facilitate operation. 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.

The Transfer Adapter is constructed of A36 carbon steel to provide an interface between the TFR and the VCC or transport cask for TSC transfer operations. The transfer adapter incorporates hydraulic cylinders which operate connector assemblies that interface with the connector welded to each of the TFR shield door assemblies. The transfer adapter interfaces with the top of the VCC using a location ring and incorporates door rails that interface with the TFR shield door rails. The transfer adapter is installed and centered on the top of the VCC and at the option of the user may be bolted to the VCC top flange. The connector assemblies are then extended to locate the female connector assemblies into a position to engage the shield door connectors as the TFR is lowered onto the transfer adapter. After the shield door locking pins are removed, the transfer adapter hydraulic system is used to retract the connector assemblies using the hydraulic cylinders and opening the shield doors. The TSC can then be lowered into the central cavity of the VCC and set down on the base plate / baffle assembly. After removal of the TSC lifting slings or lifting system, the hydraulic system is actuated to close the shield doors to allow the TFR to be removed from the top of the transfer adapter. The transfer adapter is then removed to allow installation of the VCC shield plug and lid.

3.1.1.4 Spent Nuclear Fuel (SNF) Assembly

The SNF assembly subcomponents consist of 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. Although licensed for BWR SNF assemblies, no BWR fuel has been loaded in NAC-UMS Systems in the US at an NRC licensed ISFSI.

3.1.2 Environments

3.1.2.1 NAC-UMS 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-UMS 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, 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, 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-UMS Systems are currently deployed at four nuclear plant sites: the Maine Yankee decommissioned site in Wiscasset, Maine adjacent to the Back River; Duke Energy's McGuire Nuclear Station in south central North Carolina on the southern shore of Lake Norman; Duke Energy's Catawba Nuclear Station in north central South Carolina on the southern shore of Lake Wylie; and Arizona Public Services Palo Verde Nuclear Generating Station in central Arizona west of Phoenix. The Maine Yankee (MY) site is located 12 miles from an open ocean, and experiences lower winter temperature conditions and moderate levels of rainfall and humidity. The Catawba (CNS) and McGuire (MNS) sites are in the southern eastern region of the country which experiences moderate weather conditions, rainfall and humidity. The Palo Verde Nuclear Generating Station (PVNGS) is in a high desert location and experiences higher ambient temperature conditions and below normal rainfall and humidity conditions. All four of the sites fall within the environmental conditions evaluated in the NAC-UMS System UFSAR [3.9.1.a thru o]. The 30-year average low and high temperatures for the NAC-UMS ISFSIs range from approximately 29.6°F in January to 77.9°F in July at MY, 30°F in January to 90.7°F at CNS, 29.6°F in January to 89°F in July at MNS, and 44.2°F in January to 108.3°F in July at PVNGS. (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-UMS Systems

There are six basic types of environments identified that envelope the conditions of the NAC-UMS System 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 lid are all exposed to the helium environment inside the TSC cavity. The temperatures of this gas can range from the ambient air temperature for a zero-decay heat load to a maximum of 700°F for the maximum canister heat load of 23 kW. The gas pressure in the TSC cavity is close to one atmosphere with a calculated maximum normal operating pressure of 7 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-UMS System the fully encased in steel environments include the NS-3 or NS-4-FR poured in the VCC shield plug, which is fully encased in a steel plate enclosure. In addition, the NS-4-FR and lead gamma shield bricks of the 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 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 200°F for maximum decay heat load and 100°F full solar conditions. TFR assembly embedded materials may be exposed to elevated temperatures (267°F) for short durations during fuel loading, transfer and unloading operations. During storage between loading operations, the TFR assembly temperatures will be maintained within a narrow range of “room temperature” when stored in a building or normal outside ambient condition if stored outside of the facility. The radiation levels of the embedded in metal components discussed above are significantly lower than those experienced by the sheltered air environment or helium gas environment.

In addition, 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 approximately 246°F for the design basis decay heat load of 23 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 vent 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 346°F based on the peak temperature of the TSC shell for the design-basis heat load of 23 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-UMS System 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-UMS System 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, nelson studs, underside of the liner top flange, concrete-side facing surfaces of the inlet and outlet vent structure, and the reinforcing rebar and lifting lug components 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 187°F for a decay heat load of 23 kW.

3.1.2.2.5 Air-Outdoor Environment (OD)

During NAC-UMS 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 116°F. The moisture and salinity levels to which the exterior surfaces of the VCC assembly are exposed may vary widely for the four NAC-UMS System ISFSI sites, 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 NAC-UMS System Transfer Cask (TFR) components that are typically housed indoors except for periodic exposure to outdoor air during TSC transfer operations at some sites. Indoor air describes the environment in a spent fuel building or other protective enclosure. At some NAC-UMS System ISFSIs that have completed NAC-UMS System loading operations, TFR components may be stored outdoors in a storage container or covered by a protective covering.

During fuel transfer operations, the TFR was generally stored inside a building where air temperatures and moisture levels are less variable and more controlled than those of the air-outdoor environment of the VCC assembly. Following completion of NAC-UMS System fuel loading operations, the TFR assembly may be stored outside with adequate protection from environmental extremes. In addition, stored TFR assemblies are not exposed to the elevated temperatures and radiation levels experienced by the TSC assembly and VCC assembly 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.

For purposes of the evaluation of aging effects in different environment, 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 degradation of a SSC subcomponent would be insufficient to cause a loss of function, or the relevant conditions do not exist at locations that utilize the NAC-UMS System for the aging effect to occur and propagate, then no aging management is 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-UMS System user operating experience, were considered. The evaluation was applied to identified SSC subcomponents.

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-UMS 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. NEI14-03, 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-UMS 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 3.2-4. 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, Transfer Cask (TFR), and Transfer Adapter (TFR). The NAC-UMS 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 MAPS [3.9.4] including environmental, thermal, mechanical, and irradiation-induced aging mechanisms as follows:

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

Not all these mechanisms are considered to be credible for each structure, system, and component (SSC) of the NAC-UMS 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-UMS 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-UMS System steel subcomponents are used in the VCC and TFR/Transfer Adapter SSCs, and are exposed to sheltered environments, outdoor air, indoor-outdoor air, and embedded in concrete. The exterior surfaces of NAC-UMS 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-UMS System TFR/Transfer Adapter are predominately exposed to an indoor environment, except for short periods of outdoor exposure during transfer operations at some facilities. 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 MgCl2, 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 Metallic Components Monitoring AMP and the Transfer Cask AMP and are discussed in Section 3.4. The potential for general corrosion of the VCC internal, external and embedded in concrete steel components (e.g., liner, pedestal, baseplate, Nelson studs, inlets/outlet assemblies, supplemental shield assemblies or bars, shield plug, lid, lifting lugs) is evaluated in a TLAA for the 40-year period of extended operation as discussed in Section 3.3. Sheltered and embedded in concrete steel components are shown to maintain their safety functions based on corrosion over the 40-year period of extended operation, and external components are shown to maintain their intended safety functions of the components if the coatings is maintained to prevent corrosion of the steel components. The External VCC Metallic Components Monitoring AMP identifies the general inspection requirements to verify, and correct as appropriate, external VCC metallic components coating to minimize general, and pitting and crevice corrosion.

Steel Components Exposed to Demineralized Water

Except for short durations of immersion of the NAC-UMS System TFR and the BWR TSC including steel fuel basket support disks in the spent fuel pool, there are no steel components of the NAC-UMS System exposed long-term to demineralized water. The NAC-UMS 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. The NAC-UMS System BWR TSC basket has carbon steel support disks that are electroless nickel plated during fabrication, and the spent fuel pool water is drained, and the TSC is vacuum dried and backfilled with high purity helium immediately after fuel loading and shield lid welding operations. 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-UMS 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-UMS System VCC, steel reinforcing rebar is 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, threaded lifting lug 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 Components Fully Encased (Steel) Environment

In the NAC-UMS 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(s) 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 PWR NAC-UMS System, there are no steel subcomponents exposed to a helium environment, as all PWR TSC and fuel basket steel components are stainless steel. In the BWR NAC-UMS TSC and fuel basket assembly steel components are also stainless steel except that the BWR fuel basket utilizes carbon steel fuel basket support disks coated with electroless nickel. However, there are no identified credible aging mechanisms for steel in a helium environment due to the low relative humidity and oxygen levels after vacuum drying operations and backfilling of the TSC with high purity helium. As such, general corrosion of the electroless nickel coated carbon steel exposed to helium is not considered to be significant, and therefore, aging management of steel in a helium environment is not required for the NAC-UMS 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-UMS System TFR 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, lifting lugs, etc.) are evaluated by TLAA 30013-2002 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, lifting lug threaded rebar) 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 30013-2002 and were shown to maintain their safety margins for the 40-year PEO as discussed in Section 3.3.

Steel Subcomponents Exposed to Encased Neutron-Shielding Materials Environments

In the NAC-UMS 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 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 PWR NAC-UMS System, there are no steel subcomponents exposed to a helium environment, as all PWR TSC and fuel basket steel components are stainless steel. In the BWR NAC-UMS System TSC and fuel basket assembly steel components are also stainless steel except that the BWR fuel basket utilizes carbon steel fuel basket support disks coated with electroless nickel. However, there are no identified credible aging mechanisms for steel in a helium environment due to the low relative humidity and oxygen levels after vacuum drying operations and backfilling of the TSC with high purity helium. As such, pitting and crevice corrosion of the electroless nickel coated carbon steel exposed to helium is considered to be insignificant, and therefore, aging management of steel in a helium environment is not required for the NAC-UMS 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 spent fuel 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-UMS 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 are between the bottom of the TSC, the ¼-inch stainless steel cover plate and the top of the steel 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 presence of the epoxy coating and thickness of the baffle weldment top plate (2 inch).

Other points of connection between SSC steel subcomponents and more noble materials include 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 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 and lifting lug bolting, as applicable.

Another area of potential galvanic corrosion is at the interface between the Transfer Cask steel retaining ring and the stainless-steel (ferritic) retaining ring bolting. 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.

There are no other potential areas of galvanic corrosion identified for the NAC-UMS System. As galvanic corrosion rates of the VCC lid bolts and washers and the retaining ring may be sufficient to affect component intended functions, galvanic corrosion is credible, and aging management is required during the 40-year period of extended operation. The applicable AMP proposed to evaluate this aging mechanism is AMP-3 for the VCC components involved and the Transfer Cask AMP-5. These are discussed in Section 3.4.

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 [3.9.45]. Bacteria resistant to radiation include *Micrococcus radiodurans*, which can tolerate 10 kilograys (kGy) [10^6rads] 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-UMS 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 a significant. As such, MIC of steel in concrete environments is not considered to be credible for the NAC-UMS System, and therefore, aging management is not required during the 40-year period of extended operation. There are no NAC-UMS 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-UMS 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-UMS System TFR and the BWR TSC including electroless nickel coated steel basket support disks in the spent fuel pool, there are no steel components of the NAC-UMS System exposed long term to demineralized water as the NAC-UMS 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-UMS System, there are shielding materials fully encased (FE) in steel components in the TFR and VCC shield plug. 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 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 (SCC)

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-UMS System steel bolting of VCC subcomponents and the TFR retaining ring 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-UMS 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 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-UMS System are at locations close to the fuel rods. However, there are no steel components in the PWR TSC and fuel basket, and therefore, are not applicable to this aging mechanism. The BWR fuel basket utilizes ASME SA533, Type B, Class 2 steel fuel basket support disks and these steel basket disk components are not at a sufficiently high temperature during storage operations to initiate creep. The maximum calculated temperature for the BWR steel basket report disks is 686°F and is 614°F during normal storage operations. In addition, the support disks are not under continuous stress and perform their ITS function under storage accident conditions. Therefore, aging management for creep of the BWR fuel basket steel components is not considered to be credible, 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 (Steel) Environments

NAC-UMS System steel subcomponents in the VCC and TFR/transfer adapter 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-UMS System is a static application, cyclic loading by a purely mechanical means is largely limited to NAC-UMS System TFR lifting trunnions, which are loaded each time a TSC is moved from the spent fuel pool to 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-UMS 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-UMS System are at locations close to the fuel rods. However, there are no steel components in the PWR TSC and fuel basket, and therefore, these components are not applicable to this aging mechanism. The BWR fuel basket utilizes ASME SA533, Type B, Class 2 steel fuel basket support disks and the disk temperatures are below the allowable established by the ASME Code for this material of 700°F (Code Case N-71-17) especially during the period of extended operation are not sufficient to produce degradation of mechanical properties of the steel support disks. Therefore, aging management for thermal aging of the BWR fuel basket steel components is not considered credible and is not required during the 40-year period of extended operation.

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

The temperatures of NAC-UMS 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-UMS System TSC shell show that the peak temperature is below 200°C [392°F]. The TSC shell temperatures will be significantly reduced during the initial 20-year storage period and the peak temperatures for steel subcomponents exposed to sheltered, outdoor air, and embedded environments will be below the temperature required to cause reductions in toughness during the period of extended operation. Therefore, thermal aging is not considered to be credible for these subcomponents and 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 [3.9.141]. 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-UMS System steel components in the fuel basket (BWR fuel basket support disks), 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-UMS 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-UMS System VCC lid bolting in outdoor environments is not considered to be exposed to sufficiently high temperatures to cause stress relaxation. Similarly, NAC-UMS System TFR bolting in indoor/outdoor environments is not considered to be exposed to high temperatures for enough time to cause stress relaxation. There are no NAC-UMS 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-UMS System TFR/transfer adapter exposed to air-indoor/outdoor, the TFR shield doors experience sliding contact with the TFR and transfer adapter door rails during TSC transfer operations. These SSC subcomponents are constructed of A350 LF2 low alloy steel and A36 carbon steel (transfer adapter) 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 / Transfer Adapter 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-UMS System subcomponents. The NAC-UMS System stainless steel components include the TSC shell weldment, structural and shield lids, 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 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-UMS 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:

$d = At^{-n}$ *and* n= 0.33 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 Fully Encased (Steel) 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, 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-UMS 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-UMS 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 Fully Encased (Steel) 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-UMS 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-UMS 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 [3.9.92; 3.9.93]. 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 [3.9.64; 3.9.65] 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-UMS System 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 lid are welded as an assembly in the NAC-UMS 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.

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 [3.9.130] 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 credible for the NAC-UMS 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-UMS System TSC is fabricated from 300 series stainless steels with some basket structural components fabricated from precipitation hardened stainless steels and high-strength steel. 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-UMS 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-UMS 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-UMS 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-UMS 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 TSC and fuel baskets were initially analyzed in FSAR in accordance with the rules of the ASME Code, Section III, Division 1, Subsection NB and NG, respectively for a period of 50 years. 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-UMS System TSC SSC subcomponents in the 40-year period of extended operation.

3.2.1.2.8 Thermal Aging

The microstructures of the NAC-UMS 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] [3.9.29; 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-UMS 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," [3.9.79] 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 NAC-UMS System 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-UMS System TSC, the other welded components exposed to the helium environment are the basket top and bottom weldments, 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-UMS 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 601°F per the UFSAR [3.9.1.a thru o], 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 credible, and therefore, aging management is required during the 40-year period of extended operation. The potential for thermal aging of the 17-4 PH structural disks us evaluated in a TLAA as discussed in Section 3.4. The appropriate TSC component tables have been revised to reflect that thermal aging is evaluated by a TLAA.

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

Because the peak temperatures for NAC-UMS 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-UMS 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-UMS System, stainless 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-UMS 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-UMS 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 an ITS structural function for the basket assembly. These are the only aluminum ITS components included in the NAC-UMS 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 Environment

Following vacuum drying of the NAC-UMS System TSC, there is very little residual water in the canister 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-UMS System BWR and PWR fuel baskets, the total surface area for the 0.5-inch-thick heat transfer disks is 25,000 in^2 to > 50,000 in^2, respectively. 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 Environment

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-UMS 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 the NAC-UMS System TSC basket assemblies, galvanic coupling may exist between aluminum and stainless-steel assembly components.

Aluminum Subcomponents Exposed to Helium Environment

There is very little residual water within a NAC-UMS 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-UMS 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 $0.4T_m$, where T_m 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-UMS System basket assembly does 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 NAC-UMS System 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 Environment

The highest temperatures within the NAC-UMS 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.

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-UMS 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 eight PWR) or six (BWR) tie rods. Due to the minimal applied loads, creep of nonstructural heat transfer disks will not produce significant damage during the 40-year period of extended operation and therefore, aging management is not required.

3.2.1.3.6 Fatigue

The NAC-UMS 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 nonstructural 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-UMS System SB209 6061-T651 aluminum alloy is used in the TSC fuel baskets to transfer heat.

Aluminum Subcomponents Exposed to Helium Environment

The 6061-T651 aluminum alloy 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-UMS 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-UMS System aluminum disks are not structural components thermal aging 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-UMS 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 a helium environment 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-UMS 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 the 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-UMS System both polymeric (NS-4-FR) and cementitious materials are 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-UMS System. In the NAC-UMS 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 TFR body, and in between the bottom plate and neutron shield boundary plates of the two bottom shield doors. The NS-4-FR and NS-3 materials utilized in the NAC-UMS System VCC shield plugs are also fully encased in a 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; 3.9.96; 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-UMS 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. The TLAA results conclude that boron depletion will not be a significant factor in the neutron shielding performance to affect the important to safety functions of the TFR and VCC shield plug neutron shielding during the 40-year period of extended operation.

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) [3.9.367; 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-UMS 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-UMS 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-UMS System design basis decay heat load of 23 kW is 200°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 [3.9.162; 3.9.163]. 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-UMS System TFR 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-UMS 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 TFR 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-UMS 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 [3.9.14; 3.9.20; 3.9.51; 3.9.85; 3.9.129; 3.9.142]. 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-UMS 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), [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-UMS 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-UMS 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-UMS 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 [6.5×10^4 – 6.5×10^6 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-UMS 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 0.4 T_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 sufficient 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-UMS 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 UMS 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-UMS 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-UMS 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-UMS 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, aging management of Boral neutron poison for radiation embrittlement in the NAC-UMS System TSC fuel baskets is not required during 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-UMS System storage application. Therefore, radiation embrittlement of Boral® is not considered to be credible. Consequently, aging management of Boral® neutron poison in the NAC-UMS System TSC fuel baskets is not required during the 40-year period of extended operation.

3.2.4 Concrete Overpack

The concrete overpacks for the stored canister in the NAC-UMS 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 a 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-UMS System VCC AMR Table 3.2-2. 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 [3.9.170; 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 AMP proposed for the potential impacts of freeze/thaw is the AMP for Reinforced Concrete Cask (VCC) Structures 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 as there is no exposed concrete in a sheltered environment, and therefore, aging management of concrete of the NAC-UMS 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; 3.9.222; 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-UMS 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-UMS System during the 40-year period of extended operation.

3.2.4.1.3 <u>Reaction with Aggregates</u>

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; 3.9.178]. 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-UMS System is required during the 40-year period of extended operation. The AMP proposed for the potential impacts of reactions to aggregates is the AMP for Reinforced Concrete Cask (VCC) Structures 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 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, NOx, 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 credible, and therefore, aging management of the NAC-UMS System is required during the 40-year period of extended operation. The applicable AMP proposed for the potential impacts of aggressive chemical attack is the Reinforced VCC Structure AMP as discussed in Section 3.4.

Concretes Exposed to Sheltered Environments

Regarding concrete in sheltered environments, external sources of sulfate, magnesium, and acid entering concrete are insignificant. In addition, the heat load from the fuel in the NAC-UMS 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-UMS 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 and lifting lug support embedments have been reported within the 60-year timeframe for concretes of moderate to high quality such as that achieved in the construction of NAC-UMS VCCs. The corrosion of reinforcing steel and other steel components embedded or partially embedded in concrete exposed to outdoor environments was evaluated in TLAA 30013-2002. 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-UMS 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 in the life of the concrete. As a result, shrinkage of concretes exposed to outdoor environments of the NAC-UMS System is not considered to be credible during the PEO, 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 [3.9.188]. 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-UMS System does not have similar design or operating features of the NUHOMS, leaching of calcium hydroxide in NAC-UMS 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 AMP for Reinforced Concrete Cask (VCC) Structures 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 [3.9.317]. Concrete structures have been regarded as being sound if 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 1013–1015 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-UMS 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 could 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-UMS 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 [3.9.192]. 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-UMS System under maximum decay heat load of 23 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-UMS 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-UMS 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-UMS 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-UMS 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-UMS 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 <u>Salt Scaling</u>

<u>Concretes Exposed to Air-Outdoor Environments Above the Freeze Line</u>

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-UMS System does have areas of horizontal structures on 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. Salt scaling damage is considered credible for NAC-UMS System VCC systems within the 60-year timeframe for concrete structures exposed to air-outdoor 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 AMP for Reinforced Concrete Cask (VCC) Structures 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 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-UMS 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-UMS 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-UMS 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 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-UMS 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 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) is expected to gather similar experimental confirmatory data to support the technical basis for storage of high-burnup (HBU) fuel during the first period of extended operation [3.9.298].

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 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 staff’s assessment condition of the SNF assembly at loading considered changes to the fuel pellets and the zirconium-based 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 or an alternate 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 high burnup zirconium-based cladding (i.e., average assembly burnups exceeding 45 GWd/MTU) include hydride-induced embrittlement, delayed hydride cracking, thermal and athermal (low-temperature) creep, and localized mechanical overload. In addition, the demonstration program discussed in NUREG/CR-6745 and NUREG/CR-6831 provided confirmation that hydride reorientation and creep will not compromise the configuration of low burnup fuel during the renewal period. Other aging mechanisms considered for both low and high burnup zirconium-based cladding include 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. Detailed discussions regarding each of these applicable aging mechanisms for cladding are provided in Section 3.2.5.1.

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 3.2.5.2.

3.2.5.1 Cladding Materials

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; 3.9.311]. 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™. For the NAC-UMS System, the maximum PWR SNF average burnup is limited to ≤ 60 GWd/MTU and for BWR SNF to ≤ 45 GWd/MTU. When discharged from the reactor and during wet storage, the faces of the hydride platelets are mostly oriented in the circumferential-axial direction, with a smaller fraction oriented in the radial-axial direction.

Once the SNF assemblies are removed from wet storage and loaded into a dry storage system, the TSC cavity is vacuum dried and backfilled with an inert gas. During vacuum drying, the temperature of the SNF assemblies and the temperature-dependent solubility limit of hydrogen in the cladding will also increase. As a result, some of the hydrides present in the cladding will redissolve as hydrogen. The amount of dissolved hydrogen will depend on the peak cladding temperature during the vacuum drying operations, which, per ISG-11, Revision 3 and NAC-UMS System CoC do not exceed 400°C [752°F] for HBU fuel. For example, the maximum dissolved hydrogen at 400°C [752°F] is approximately 200 wppm based on representative solubility correlations [3.9.320; 3.9.328]. Once the loaded TSC is dried and backfilled, the cladding temperature will decrease over time, and upon a sufficient temperature drop (~65°C [117°F]), some of the hydrogen in solution will reprecipitate as new hydrides. During this process, the orientation of these precipitated hydrides may change from the circumferential to the radial-axial direction. The degree of reorientation is driven by the metallurgical microstructure of the cladding alloy and the cladding hoop stresses during drying operations and subsequent cooling, which are determined by the rod internal pressure at a given gas temperature.

The primary driving force for radial hydride reorientation is the cladding hoop stresses, which are determined by the peak cladding temperature during drying operations. A review performed by staff indicates that there is no consensus in the literature on minimum level or threshold hoop stresses needed to reorient hydrides for a given cladding alloy and temperature, as discussed in the following references:

- *Zircaloy-4:* Data from Chung [3.9.289], Daum et al. [3.9.295], and Chu et al. [3.9.288] suggest that the threshold hoop stress for hydride reorientation in Zircaloy-4 is about 90 MPa [13 ksi] for peak temperatures at or near 400°C [752°F] for both irradiated and unirradiated rods. Other data obtained from irradiated cladding [3.9.297; 3.9.284; 3.9.304] suggest that hoop stresses greater than 120 MPa [17 ksi] may be required. Most recently, Kim et al [3.9.324] showed threshold stresses for hydride reorientation in unirradiated Zircaloy-4 of 60 ± 5 MPa [8.7 ± 0.7 ksi] at 400° C [752°F], 68 ± 5 MPa [9.8 ± 0.7 ksi] at 335°C [635°F], 75 ± 6 MPa [10.9 ± 0.9 ksi] at 300°C [572°F], and 90 ± 6 MPa [13.0 ± 0.9 ksi] at 235°C [455°F]. Kamimura [3.9.319] also reported a threshold stress for Zircaloy-4 of about 100 MPa [16 ksi] at 275°C [527°F] for a nominal burnup of 48 GWd/MTU.
- *Zircaloy-2:* Kamimura [3.9.319] reported a threshold hoop stress of 70 MPa [10 ksi] for Zircaloy-2 (no zirconium liner) of nominal burnup of 40 GWd/MTU at 200°C [392°F], and 70 MPa [10 ksi] for Zircaloy-2 (with zirconium liner) of nominal 50 GWd/MTU and 55 GWd/MTU burnups at 300°C [572°F].
- *Advanced alloys:* Kamimura [3.9.319] reported a threshold stress of 90 MPa [13 ksi] for ZIRLO™ at 250°C [482°F] for a nominal burnup of 55 GWd/MTU. Billone et al. [3.9.280] reported reorientation of M5® cladding at their lowest studied hoop stress of 90 MPa [16 ksi] for a peak cladding temperature of 400° [752°F] and nominal burnup of 68 GWd/MTU.

These threshold hoop stresses for hydride reorientation were compared to estimated hoop stresses for representative BWR and PWR fuel assemblies. Raynaud and Einziger [3.9.343] estimated the hoop stresses for 10x10 BWR and 17x17 PWR fuel assemblies as a function of decay gas release and fuel pellet swelling, which accounted for decay gas released to the pellet clad gap. The maximum calculated hoop stress during drying operations for the BWR cladding was approximately 40 MPa [5.8 ksi] at a peak cladding temperature close to 400°C [752°F]. Similarly, the maximum calculated hoop stress during drying operations for PWR cladding was approximately 100 MPa

[14.5 ksi] at 400°C [752°F], which rapidly decays and falls well below 50 MPa after a few decades in dry storage. These calculations did not account for ZIRLO™-clad integral fuel burnable absorber (IFBA) rods with hollow and solid blanket pellets; however, these rods are expected to experience higher maximum hoop stresses [3.9.281]. Since the calculated hoop stresses exceed the experimental values in the literature for when radial hydride reorientation was observed, the staff considers that the radial hydride precipitation is credible in both in BWR and PWR fuel claddings in dry storage.

The cladding alloy and corresponding fabrication process are also important factors for defining the extent of hydride reorientation. Two predominant cladding microstructures are produced during fabrication: (1) recrystallized annealed (RXA) and (2) cold worked stress relieved (CWSR) annealed. Zircaloy-4 (PWR) and ZIRLO™(PWR) are generally CWSR, whereas Zircaloy-2 and M5® are RXA. RXA claddings are more susceptible to hydride reorientation because hydrides tend to precipitate in the grain boundaries and these cladding types have a larger fraction of grain boundaries in the radial direction (equiaxed grains) relative to CWSR claddings (which have more elongated grains). However, it is important to note that RXA claddings have lower hydrogen contents following reactor irradiation and, therefore, a lower overall concentration of hydrides.

The Staff also considered the effect of the cladding cooling rate on the degree of hydride reorientation. The cooling rate post-drying and under dry storage is expected to be in the range of 10-3 to 10-5 degrees C/hr [1.8 x 10-3 to 1.8 x 10-5 degrees F/hr]. Most of the experimental studies reported in the literature have used cooling rates in the range of 0.6–30 degrees C/hr [1.08–54 degrees F/hr] [3.9.276]. However, an analysis of ductility data collected at different cooling rates in [3.9.276] does not show a clear trend. Chan [3.9.286] also developed a micromechanical model to determine the effect of slow cooling rates on hydride reorientation and morphology, including volume fraction of both radial and circumferential hydrides and continuity of the hydride network. Using experimental data to validate the model, Chan concluded that the cooling rate exerts no direct influence on radial hydride precipitation; instead, hydride orientation is dictated by the cladding stresses during hydride precipitation, regardless of the cooling rate. Therefore, the staff concludes that the slow cooling rates experienced post-drying and during dry storage are not expected to inhibit the precipitation of radial hydrides.

Cladding with a high concentration of radial hydrides (as determined by the DSS drying conditions) has been shown to have reduced ductility under pinch-load stresses at sufficiently low temperatures, thereby affecting the ability to retrieve the HBU fuel [3.9.280; 3.9.276]. The degradation of the mechanical properties at a particular temperature (described as the “ductile-to-brittle transition temperature” or DBTT) depends on the interconnectivity and number density of radial hydrides (as determined by their length, distribution, and orientation), and the thickness of the outer-surface hydride rim. This formation of a rim of radial hydrides has led the staff to express concern about potential cladding failures when subjected to pinch-load stresses higher than the fuel’s mechanical limit, if the cladding temperature decreases below the corresponding DBTT [3.9.338]. Therefore, as the cladding cools down during the 60-year timeframe, the extent of radial hydride reorientation and the DBTT have been considered important for evaluating the cladding performance and ensuring that the HBU fuel remains in the analyzed configuration.

Available DBTT data on HBU fuel cladding samples with radial hydrides have been obtained under conservative conditions and acceptance criteria (e.g., testing was performed on defueled samples, which do not account for the composite pellet-clad mechanical behavior) ([3.9.302], [3.9.280; 3.9.276]). For example, Billone et al. [3.9.280] showed that Zircaloy-4, ZIRLO™, and M5® cladding samples subjected to a radial hydride reorientation treatment exhibited lower ductility under pinch-load stresses at low relative temperatures {less than 200 degrees C [392 degrees F]}. The radial hydride treatment was designed to simulate drying and storage conditions {i.e., peak cladding temperature of 400 degrees C [752 degrees F] and peak hoop stresses of ~110 MPa [16.0 ksi] and ~140 MPa [20.3 ksi]}. General conclusions from Billone et al. [3.9.280] were that: (1) the DBTT generally increases with increasing hoop stresses (i.e., the degradation of cladding mechanical properties shifts to higher cladding temperature), (2) both the susceptibility to radial hydride precipitation and degradation of mechanical properties depend on cladding type and initial hydrogen content, and (3) depending on the cladding and test conditions, the DBTT can occur at temperatures in the range of approximately 20 degrees C to 185 degrees C [68 to 328 degrees F]. The results for as-irradiated Zircaloy-4 are consistent with studies by [3.9.363].

The staff has long expected that hydride reorientation would not compromise HBU fuel cladding integrity as a result of fuel rod bending experienced during dry storage operations, as the principal tensile stress field associated with rod bending caused by lateral inertia loads is parallel to both radial and circumferential hydrides [3.9.152]. The staff has considered that any reduced cladding ductility resulting from hydride reorientation could only potentially compromise the analyzed fuel configuration for pinch loads experienced during design-basis drop accidents (i.e., postulated drops during the removal or transfer of a canister or cask retrieval at the end of storage operations, as described in the approved design bases). Pinch loads could occur because of rod-to-grid spacer contact, rod-to-rod contact, or rod-to-basket contact during the drop accident. If the fuel temperature were sufficiently low at the time of the accident, these pinch loads could compromise the analyzed fuel configuration.

The NRC has since sponsored confirmatory research at Oak Ridge National Laboratory, as discussed in NUREG/CR-7198, Revision 1, “Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications” [3.9.360], to obtain results to compare to the staff’s expectations. The research discussed in NUREG/CR-7198, Revision 1, provided results on both the static bending response and the fatigue strength of HBU fuel rods when considered as a composite system of cladding and fuel pellets. These results have allowed the staff to conduct a more accurate engineering assessment of the structural behavior of the composite fuel rod system during dry storage (i.e., the structural support imparted by the fuel pellet). The staff has documented this assessment in a separate technical report, NUREG–2224, “Dry Storage and Transportation of High Burnup Spent Nuclear Fuel—Draft Report for Comment,” [3.9.368], which provides a technical basis supporting the conclusion that hydride reorientation is inconsequential to the expected loads during design-basis drop accidents in storage and during seismic loading conditions.

Considering the hydrogen content, peak drying temperatures, and corresponding hoop stresses, the staff has concluded that hydride reorientation in zirconium-based HBU cladding is credible during the 60-year timeframe. However, in accordance with the technical bases discussed in NUREG–2224, hydride reorientation is not expected to result in cladding failures and reconfiguration of the fuel, if the approved design bases are consistent with the acceptance criteria in ISG-11, Revision 3. The staff also recognizes that the experimental evidence used in support of the conclusions in NUREG–2224 is based on short-term testing. Therefore, the staff considers it prudent that evidence from HBU fuel in dry storage beyond 20 years be gathered in the field and reviewed. One acceptable approach relies on the evaluation of data from a demonstration (surrogate) program consistent with the guidance in Appendix D to NUREG–1927, Revision 1 [3.9.2]. For example, destructive examination from the DOE/EPRI cask demonstration project [3.9.298] may be used as confirmation that hydride reorientation has not compromised the assumed cladding mechanical properties in accordance with the approved design bases

For the NAC-UMS System CoC renewal for uncanned HBU fuel, the destructive examinations from the DOE/EPRI cask demonstration project [3.9.298] are planned to be used as confirmation that hydride-induced embrittlement has not compromised the ability to retrieve the HBU spent fuel on a single-assembly basis. The aging management proposed for investigating potential hydride re-orientation is presented in the High Burnup Fuel Monitoring and Assessment AMP and is discussed in Section 3.4.

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 [3.9.291]. 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 [3.9.327; 3.9.322] and representative of dry storage [3.9.349; 3.9.6].

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. [3.9.290] were able to induce DHC in Zircaloy-4 at 200 wppm of hydrogen. Regarding requisite existing (incipient) crack tips, EPRI [3.9.366] 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 17 x 17 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 [3.9.343] 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 [3.9.286] 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] [3.9.286, Figures 2 and 3]. Kubo et al. [3.9.327] also compiled K_{IH} values for Zircaloy-2 in the range of 3–13 MPa√m [2.73–11.8 ksi√in]. Kim [3.9.321] 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 [3.9.343] 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 KI_{H} 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 in Raynaud and Einziger [3.9.343] did not account for the hoop stresses in ZIRLO™-clad IFBA rods with hollow and solid blanket pellets, which are expected to be higher than standard rods [3.9.281]. Therefore, the staff performed similar calculations to those in Raynaud and Einziger [3.9.343] for IFBA rods, assuming a K_{IH} value of 5 MPa√m [2.73 ksi√in] and a conservative IFBA-rod hoop stress of 130 MPa [21.75 ksi]. These calculations 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, the staff has 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-UMS cladding temperatures design-bases peak cladding temperature are 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. Therefore, the assumptions and analyses discussed above are considered reasonably bounding and indicate that DHC is not a credible aging mechanism during the 60-year timeframe.

The staff also considered a DHC model proposed by Kim [3.9.323, 3.9.222], which evaluated cladding absent thermal cycling, where multiple parameters were analyzed, including creep deformation, cladding burnup, temperatures of hydride formation and dissolution (solvus hysteresis), and the δ-to-γ (face-centered cubic to faced-centered tetragonal) hydride phase transition. This model, still under review by the international DHC research community, suggests that K_{IH} may be reduced (i) upon cooling below 180 degrees C [356 degrees F] (due to a hydride phase transformation from the γ to δ phase) and (ii) if there are sufficient stresses and stress risers in the rod (e.g., residual stresses at the end cap weld region, incipient cracks due to fuel-cladding interaction). Thermal gradients may also affect the kinetics of hydride precipitation. The staff reviewed this study, in light of the assumptions made in the previous discussion. However, Kim [3.9.323, 3.9.222] does not quantify K_{IH} values; therefore, adequate conclusions cannot be made with respect to threshold stresses. The NRC [3.9.23] and Hanson et al [3.9.85] summarized Kim's [3.9.323, 3.9.222] work and proposed additional research for confirmation.

Finally, the staff considered the contribution of cladding stresses due to pellet-clad bonding and its potential to facilitate DHC initiation [3.9.361, 3.9.362]. The previously-discussed Raynaud and Einziger [3.9.343] 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 pellet-to-cladding mechanical interaction (PCMI) than for a perfectly cylindrical pellet (as assumed in the paper). Ahn et al. [3.9.274] estimated stress concentrations from pellet-clad mechanical stresses due to the radiation-induced pellet swelling up to 100 years, independent of hoop stresses due to continued 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. For low burnup fuel, pellet expansion stresses will be minimal, because the gap between the cladding and the pellet will accommodate the swelling. Literature indicates that radiation-induced pellet swelling is expected to reach its maximum value beyond the 60-year timeframe [3.9.345, 3.9.346, 3.9.347]. Therefore, the Staff does not have evidence that the potential for high PCMI stress concentrations due to radiation-induced pellet swelling would facilitate DHC crack initiation until past the first renewal period.

Based on the staff analysis in MAPS, it has been determined that DHC is not a credible aging mechanism for the NAC-UMS 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 main driving force for cladding creep at a given temperature is the hoop stress caused by internal rod pressure, which accounts for the fission and decay gases released to the interspace between the fuel and cladding. Fuel pellet swelling also may result in localized stresses due to the mechanical interaction between the cladding and the fuel. Pellet swelling may occur due to (i) the incorporation of soluble and insoluble solid fission products in the fuel matrix, (ii) the formation of intra- and intergranular fission gas bubbles, particularly in the hot interior region of a fuel pellet, and (iii) the formation of a large number of small gas bubbles in the fine-grained ceramic structure that builds inward from the outer pellet surface for HBU fuel.

Raynaud and Einziger [3.9.343] estimated the transient cladding hoop stresses during dry storage for typical 10×10 BWR and 17×17 PWR fuel assemblies. 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 study reported peak cladding hoop stresses less than 50 MPa [7.25 ksi] for BWR and less than 100 MPa [14.5 ksi] for PWR fuel assemblies. Raynaud and Einziger used these hoop stress estimates to calculate cumulative cladding strains for the representative assemblies over a 60-year period of dry storage. The authors reported a maximum cladding strain of 0.54 percent for the representative 10×10 BWR fuel cladding and 1.04 percent for the representative 17×17 PWR fuel cladding. However, these calculations did not account for the hoop stresses in ZIRLO™-clad IFBA rods with hollow and solid blanket pellets, which are expected to be higher than those for standard rods [3.9.281]. Therefore, the staff performed calculations to estimate the cladding strain for IFBA rods using the Raynaud and Einziger approach. Using a conservatively bounding hoop stress of 150 MPa [21.75 ksi], the maximum cladding strain was estimated to be near 2.1 percent. The elastic strain limit for various zirconium-based cladding alloys with circumferential hydrides is less than 1 percent [3.9.306] and is expected to be lower for cladding containing both circumferential and radial hydrides. Therefore, the staff concludes that the cladding in both standard and IFBA fuel rods is expected to undergo creep during the 60-year timeframe.

ISG-11, Revision 3 [3.9.10] includes acceptance criteria (regarding maximum fuel clad temperature during dry storage operations and adequate thermal cycling limits) to provide reasonable assurance that the spent fuel assemblies will remain in the configuration analyzed in the approved design bases. The references cited in ISG-11, Revision 3, provide experimental evidence that cladding failures are not expected for creep strains below 2 percent. These references provide support that gross ruptures of the cladding are unlikely due to creep during dry storage, because the creep-induced strain is expected to be near or less than 2 percent for the majority of the cladding alloys and close to 2 percent for the ZIRLO™-clad IFBA rods. For example, no failures were observed for creep strains below 2 percent strain for in-creep tests at temperatures between 250°C and 400°C [482°F and 752°F] for Zircaloy cladding irradiated up to burnup of 64 GWd/MTU [3.9.355; 3.9.304; 3.9.273]. Gross ruptures of the cladding were observed after creep strains exceeded 8 percent [3.9.283]. Recent data on optimized ZIRLO™ [3.9.342] indicate a plastic strain range in the same range as Zircaloy. In addition, Bouffioux and Rupa [3.9.283] conducted various cladding creep tests with unirradiated, prehydrided, stress-relief annealed low-Sn Zircaloy-4 PWR cladding tubes, with hydrogen levels in the range of 100–1,100 wppm. The authors observed gross ruptures of the cladding only after creep strains exceeding 8 percent. Tsai and Billone [3.9.358] also tested irradiated stress-relief annealed Zircaloy-4 with varying levels of hydrogen levels at various temperature and hoop stresses, which did not reveal cladding failures at a strain of 5.83 percent. More recent data on optimized ZIRLO™ by Pan et al. [3.9.342] also indicate a plastic strain range in the same range as Zircaloy.

It has been determined by the staff that thermal creep of zirconium-based cladding is credible during the 60-year timeframe. However, as the NAC-UMS System design bases is consistent with the acceptance criteria of ISG-11, Revision 3, the high creep capacity of zirconium-based alloys will preclude resultant cladding failures and reconfiguration of the fuel. It is recognized that the experimental evidence used in support of ISG-11, Revision 3, is based on short-term testing and issued ISG-24 [3.9.23]. Therefore, the staff issued guidance in Appendix D of NUREG–1927, Revision 1 [3.9.2] for the use of a demonstration program to confirm these expected fuel conditions after a substantial storage period (~10 years). The program would provide confirmation for accelerated cladding creep testing used as basis for the guidance recommendation for the maximum temperature in ISG-11 [3.9.10], and that sufficient creep capacity exists for the renewal period. For example, nondestructive and destructive examination from the DOE/EPRI cask demonstration project [3.9.298] may be used as confirmation that the design-basis fuel remains in the analyzed configuration and that sufficient creep margin exists for the first renewal period.

For the NAC-UMS System CoC renewal for uncanned HBU fuel, the destructive examinations from the DOE/EPRI cask demonstration project [3.9.298] are planned to be used as confirmation that thermal creep has not compromised the ability to retrieve the spent fuel on a single-assembly basis. The AMP proposed for investigating potential for thermal creep in HBU is presented in the High Burnup Fuel Monitoring and Assessment AMP as discussed in Section 3.4.

3.2.5.1.4 <u>Low-Temperature Creep</u>

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) [3.9.336], 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 [3.9.315]. Since both titanium and zirconium have the same crystalline structure (hexagonal close packed crystalline), the zirconium-based cladding is 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 [3.9.275] reported that 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]; [3.9.306; 3.9.301; 3.9.285]) 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]) [3.9.312]. 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 [3.9.343] 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] [3.9.284]. 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 [3.9.306; 3.9.301; 3.9.285]. Further, more recent data [3.9.324; 3.9.325] 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 staff further considered the contribution of cladding stresses due to pellet-clad bonding and its potential to facilitate athermal creep. The previously discussed Raynaud and Einziger study [3.9.343] 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 [3.9.345; 3.9.346; 3.9.347]. 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-UMS System for low-temperature creep is not required during the 40-year period of extended operation.

3.2.5.1.5 Mechanical Overload (high burnup fuel)

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 (nonthrough-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].

A method previously used to characterize PCMI failures in the cladding involves measuring the creep strain capacity at a given creep strain rate [3.9.316]. More specifically, PCMI-induced failures are observed when the cladding strain at a given strain rate exceeds a threshold [3.9.316, 3.9.302]. The threshold strain is a function of cladding temperature, irradiation, and hydrogen concentration. PCMI-induced failures have been reported at cladding strains exceeding 1 percent for strain rates in the range of 10^{-5} to 10^{-3} s^{-1} at room temperature for various levels of hydrogen concentration [3.9.316] . At higher temperatures, the strain at failure is above 6 percent between 523 and 673 K [482 to 752 degrees F] for strain rates in the range of 10^{-5} to $10^{-3}s^{-1}$ [3.9.316] . This threshold strain at higher temperature is applicable for cladding hydrogen content up to 1,200 wppm. These results are consistent with those by Fuketa et al. [3.9.302], which exhibited similar threshold strains between 373 and 573 K [212 to 572 degrees F] with hydrogen concentrations up to 1,450 wppm. These results can be compared with data discussed in Section 3.5.1.3, which show that, for comparable strain rates in the order of 10^{-4} s^{-1} to 10^{-5} s^{-1}, no failures were observed for creep strains below 2 percent for in-creep tests at temperatures between 150 and 400 degrees C [423 and 752 degrees F] for Zircaloy cladding irradiated up to burnup of 64 GWd/MtU [3.5.355; 3.9.304; 3.9.367].

The staff reviewed the aforementioned creep strain and strain rate threshold criteria against the results in Raynaud and Einziger [3.9.343], which estimated the temperature-dependent hoop stresses on the cladding while accounting for credible release of fission and decay gases and pellet swelling. Raynaud and Einziger estimated maximum cladding strains of 0.54 percent for the 10×10 BWR fuel cladding and 1.04 percent for the 17×17 PWR fuel cladding at a strain rate of 10^{-10} s^{-1} expected during dry storage. The authors stated that all the cladding strain is expected to occur during the first 50 years of storage. These calculations did not account for the hoop stresses in ZIRLO™-clad IFBA rods with hollow and solid blanket pellets, which are expected to be higher than standard rods [3.9.281].

The Staff performed calculations to estimate the cladding strain for IFBA rods using the Raynaud and Einziger [3.9.343] approach. Using a conservatively bounding hoop stress of 150 MPa [21.75 ksi], the maximum cladding strain was estimated to be near 2.1 percent for IFBA rods. These values indicate sufficient strain capacity per the previously discussed creep strain and strain rate threshold criteria [3.5.355; 3.9.304] , which is considered conservatively bounding as the strain rates in dry storage are expected to be approximately five to seven orders of magnitude lower than 10^{-5} to 10^{-3} s^{-1}.

Due to low levels of creep strain, strain rate, and temperature-dependent hoop stresses experienced during NAC-UMS 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: $Zr + 2H_2O = ZrO_2 + 2H_2$ [3.9.94; 3.9.291; 3.9.348]. Various scoping calculations were performed [3.9.94] 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 insignificant, and aging management for cladding oxidation in the NAC-UMS 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 [3.9.341]. Zirconium is a passive material and is protected by a ZrO_2 surface film [3.9.341]. 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. For example, pitting of zirconium has been shown to occur in hydrochloric acid solutions containing ferric (Fe^{3+}) or cupric (Cu^{2+}) ions [3.9.341].

Inside the NAC-UMS 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. 94]. 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-UMS 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-UMS 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 standard electrode potential for zirconium and ZrO_2 in aqueous solution at 25 degrees C [77 degrees F] is approximately in the range of −1.5 to −1.6 V_{SHE}, where the subscript "SHE" stands for standard hydrogen electrode [3.9.309]. The standard electrode potentials for chromium, nickel, molybdenum, and iron are approximately equal to −0.74, −0.20, −0.26, and −0.44 V_{SHE}, respectively, at 25 degrees C [77 degrees F] [3.9.278; [3.9.309]. The standard electrode potential data indicate that zirconium would be oxidized to zirconium ions during the galvanic reaction, and oxidizing species, such as oxygen and hydrogen peroxide in aqueous solution, would be reduced at the Inconel alloy.

The extent of loss of cladding material would depend on the amount of oxidants present in the condensed water. For example, per the stoichiometry of the oxidation and reduction reactions [3.9.94], reduction of 1 mole of hydrogen peroxide would result in oxidation of 0.5 mole of zirconium. Similarly, reduction of 1 mole of oxygen would result in oxidation of 1.0 mole of zirconium. Jung et al. [3.9.94] reported scoping calculations to determine the extent of zirconium oxidation with 1 mole of a 5-weight percent H_2O_2 aqueous solution saturated with oxygen at 25 degrees C [77 degrees F] and 1 atm [14.7 psi]. Jung et al. [3.9.94] concluded that the extent of oxidation would depend on the spread of the condensed water over the large surface area. Therefore, the effect of galvanic corrosion is not expected to be localized.

The amount of residual water inside the NAC-UMS System 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-UMS 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 [3.9.363; 3.9.353]. 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 [3.9.340; 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.

Prescatore and Cowgill [3.9.300] compiled SCC failure data from Yagee et al. [3.9.365; 3.9.366], Mattas et al. [3.9.332], Shimada and Nagai [3.9.351], Kreyns et al. [3.9.329], and Crescimanno 3.9.294] for the following irradiated cladding materials:

- recrystallized Zircaloy-2,
- stress-relieved Zircaloy-2,
- recrystallized Zircaloy-4, and
- stress-relived Zircaloy-4.

For Zircaloy-2, the reported data's temperature and tensile stress ranges were 325 to 350 degrees C [617 to 662 degrees F], and 119 to 513 MPa [17.3 to 74.4 ksi], respectively. Similarly, for

Zircaloy-4, the reported SCC data's temperature and tensile stress ranges were 316 to 350 degrees C [601 to 662 degrees F], and 164 to 414 MPa [23.8 to 60 ksi], respectively. The SCC-induced failure was reported at 157 MPa [22.8 ksi] and 325 degrees C [617 degrees F] for Zircaloy-2, and at 205 MPa [29.7 ksi] and 360 degrees C [680 degrees F] for Zircaloy-4 [3.9.366].

Regarding the two failure data points, Prescatore and Cowgill [3.9.300] argued that failures were misclassified as SCC-induced failures and were more akin to nondetrimental pinhole breaches. Prescatore and Cowgill stated that gross rupture, in the form of axial splitting, was noted in many instances when the stress was greater than about 270 MPa [39.2 ksi], but at lower stresses, pinhole leakage was by far the more common failure mode. If the 157 MPa [22.8 ksi] failure data point is excluded from the data for Zircaloy-2, as argued by Prescatore and Cowgill, the next incident of the SCC-induced failure is noted at 247 MPa [35.8 ksi]. Similarly, if the 205 MPa [29.7 ksi] failure data point is excluded for Zircaloy-4, as argued by Prescatore and Cowgill, the next incident of the SCC-induced failure is noted at 273 MPa [39.6 ksi]. This analysis indicates that at least 240 MPa [34.8 ksi] of hoop stresses are needed to induce SCC for both Zircaloy-2 and Zircaloy-4.

Recent work by Raynaud and Einziger [3.9.343] shows that hoop stresses are expected to be below 100 MPa [14.5 ksi], with the most realistic estimate of release of the decay and fission gases from fuel pellets and with the best estimate of fuel swelling during a 300-year dry storage period. However, hoop stresses in ZIRLO™-clad IFBA rods with hollow and solid blanket pellets could be considerably higher. The Raynaud and Einziger [3.9.343] 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 Raynaud and Einziger [3.9.343]].

Ahn et al. [3.9.274] estimated stress concentrations from pellet-clad mechanical stresses due to 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. For low-burnup fuel, pellet expansion stresses will be minimal, because the gap between the cladding and the pellet will accommodate the swelling. Literature indicates that radiation- induced pellet swelling is expected to reach its maximum beyond the first renewal period [3.9.345; 3.9.346; 3.9.347].

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-UMS 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 [3.9.85; 3.920]. Embrittlement is largely observed during reactor operation due to cumulative fast neutron fluence on the order of 10^{22} n/cm^2 [6.5 × 10^{22}

n/in^2] [3.9.310]for recrystallized annealed Zircaloy-2 and cold-worked stress-relieved Zircaloy-4 [3.9.335]. During normal operation in the reactor, the cladding material is bombarded with fast neutrons that cause atomic displacement cascades, resulting in the formation of point defects [3.9.85; 3.920; 3.9.129]. This leads to the reduction in the mechanical properties of the cladding material. The staff evaluates these effects in its review of initial licenses to store spent fuel. To date, the staff has concluded that the changes in properties during reactor operation do not prevent the cladding from fulfilling its intended functions during the initial dry storage term.

In dry storage, the cumulative neutron fluence is expected be five orders of magnitude less than in reactor service [3.9.94]. In addition, annealing of irradiation hardening could occur during storage, which would help recover some ductility. It has been shown in literature [3.9.334; 3.9.357] that a post-irradiation heat treatment performed at a temperature above the irradiation temperature can lead to the recovery of the radiation-induced hardening and increased ductility of the cladding. Ito et al. [3.9.314] further showed that hardness also recovers at temperatures lower than an irradiation temperature of 360 degrees C [680 degrees F]. More specifically, Ito et al. [3.9.314] showed that hardness continued to recover, albeit quite slowly, at temperatures as low as 330 degrees C [626 degrees F] for 8,000 hours (0.9 year), and nearly 50 percent recovery was observed compared to the annealing over the same time at 360 degrees C [680 degrees F]. Thus, over many years of extended storage, it is possible that thermal annealing could increase cladding ductility, thereby reducing the effects of radiation embrittlement.

As stated above, the Staff has concluded that irradiation during reactor operation does not reduce cladding properties to an extent that prevents the cladding from fulfilling its intended functions. Also, the effects of additional irradiation during dry storage (five orders of magnitude less than in reactor service) are expected to be negligible. As a result, radiation embrittlement of cladding is not considered credible in the NAC-UMS System, and therefore, aging management is not required during the 60-year timeframe

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.

Devoe and Robb [3.9.296] conducted steady-state analyses to show that the change in peak cladding temperature is directly proportional to the change in external air temperature of the canister. Although the large thermal mass of the DSS is likely to reduce the amplitude and frequency of the thermal cycles on fuel and cladding temperature, Devoe and Robb [3.9.296] assumed a correlation coefficient of unity between the peak cladding and external air temperature. Thus, a 1 degree C [1.8-degree F] change in air temperature would result in approximately 1 degree C [1.8 degree F] change in cladding temperature. When evaluating daily temperature fluctuations, the analysis assumed a conservative 25 degrees C [77 degrees F] maximum daily change [equivalent to 45 degrees F change], which is the mean daily temperature change in the United States. The model further assumes a total of 21,900 thermal cycles, corresponding to steady-state temperature cycle every day for 60 years. The staff assumed these conditions to determine if the resulting changes in cladding hoop stresses could lead to fatigue-induced failure of the cladding.

Raynaud and Einziger [3.9.343] estimated the cladding hoop stresses while accounting for release of fission gases and decay gases during storage, including pellet swelling stresses due to radiation damage during storage. Raynaud and Einziger estimates included the effect of fuel temperature on cladding hoop stresses. As per the Raynaud and Einziger estimates, a 25 degree C [77 degree F] variation [45 degree F variation] in cladding temperature will cause up to 10 and 30 MPa [1.45 and 4.35 ksi] fluctuations in hoop stress of the BWR and PWR claddings, respectively.

Lin and Haicheng [3.9.330] conducted experimental studies to determine fatigue properties of zirconium and Zircaloy-4. Lin and Haicheng provided a fatigue lifetime curve for zirconium and Zircaloy-4 under reversal bending as a function of the cyclic stress. As per the fatigue lifetime curve in Lin and Haicheng, a cyclic stress amplitude of more than 260 MPa [37.7 ksi] is needed for fatigue-induced failure in Zircaloy-4 in 10^7 cycles. The curve also bounds the data for zirconium, and hence, is also assumed to be applicable for other zirconium- based cladding materials, such as Zircaloy-2, ZIRLO™, and M5®. Therefore, using the fatigue lifetime curve in Lin and Haicheng, these fluctuations in hoop stresses (per the assumed conditions in [3.9.296]) are not sufficient for fatigue-induced failure in the cladding.

The Staff also evaluated the effects of extreme seasonal temperature variations, as these are expected to be significantly higher than daily variations and could result in higher cyclic stress amplitudes. Using the off-normal DSS operating conditions of −40 degrees C [−40 degrees F] (winter) and 103 degrees C [217 degrees F] (summer) yields a maximum seasonal temperature variation of 143 degrees C [variation of 257 degrees F]. Similar to the previous analysis, per the Raynaud and Einziger estimates, a 143 degree C variation [257.4 degree F variation] in cladding temperature will cause up to 10 and 55 MPa [1.45 and 7.8 ksi] fluctuations in hoop stress of the BWR and PWR claddings, respectively. Using the fatigue lifetime curve in Lin and Haicheng, [3.9.330] these fluctuations in hoop stresses (per the assumed conditions in Devoe and Robb [3.9.296]) are also not sufficient for fatigue-induced failure in the cladding.

As discussed in Section 3.2.1.1.7, the cyclic stress, σ, induced by the thermal variations also depends on 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 component. Since the degree of constraint for the cladding is not readily available for cladding, a conservative approach is employed to estimate the cyclic stresses and associated potential impact of thermal fatigue. The coefficient of thermal expansion is estimated to be approximately 4.16×10^{-6}/K, based on the data in Luscher and Geelhood [3.9.331]. The Young's modulus of elasticity of various zirconium-based cladding materials ranges between 32 and 100 GPa [4,641 and 14,504 ksi] [3.9.331]; a value of 100 GPa [14,504 ksi] is conservatively used. The assumed values of α_0 and E result in a thermally induced cyclic stress of 10.4 MPa [1.5 ksi] and 59.5 MPa [8.6 ksi] for ΔT equal to 25 and 143 degrees C [45 and 257 degrees F], respectively. As per the fatigue lifetime curve in Lin and Haicheng [3.9.330], these fluctuations in hoop stresses are also not sufficient for fatigue-induced failure in the cladding.

The Staff further considered the cumulative cyclic stresses for all cases described above, which results 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, per Lin and Haicheng [3.9.330]. Therefore, the Staff concluded that fatigue-induced failure of the cladding is not credible during the 60-year timeframe, and aging management for fatigue-induced failure of the cladding of spent fuel stored in the NAC-UMS System is not required

3.2.5.2 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.

3.2.5.2.1 Creep

Creep is defined as the time-dependent deformation that takes place at an elevated temperature and constant stress. Because the deformation processes that produce creep are thermally activated, the rate of this time-dependent deformation (i.e., the creep rate) is a strong function of the temperature. The creep rate also depends on the applied stress but does not generally vary with the environment. As a general rule of thumb, at temperatures below $0.4T_m$, where T_m is the melting point of the metal in Kelvin, thermal activation is insufficient to produce significant creep [3.9.46]. The melting temperature of various zirconium alloys is above 1,800 degrees C [3,272 degrees F]. Similarly, the melting temperature of various Inconel alloys is above 1,260 degrees C [2,300 degrees F]. In addition, the melting temperature of 304L stainless steels is close to 1,400 degrees C [2,552 degrees F].

Regarding the zirconium alloys, the $0.4\ T_m$ criterion yields a creep threshold of 556 degrees C [1,033 degrees F]. The maximum expected temperature of fuel cladding has been estimated to be 400 degrees C [752 degrees F] at the beginning of storage [3.9.94]. This cladding temperature is expected to decrease to around 266 degrees C [510 degrees F] after 20 years and to approximately 127 degrees C [261 degrees F] after 60 years. This indicates that creep of the zirconium alloys is unlikely during the renewal period.

Regarding Inconel alloys, the $0.4T_m$ criterion yields a creep threshold of 340 degrees C [644 degrees F]. As stated previously, the peak temperature inside the storage canister is expected to be below 266 degrees C [510 degrees F] after 20 years of storage. This indicates that creep of various Inconel alloys is unlikely during the renewal period.

Regarding 304L stainless steel, the 0.4 T_m criterion yields a creep threshold of 396 degrees C [755 degrees F]. As stated previously, the peak temperature inside the storage canister is expected to be below 300 degrees C [572 degrees F] after 20 years of storage. Further, the 0.4 T_m rule of thumb underestimates the minimum creep temperature for steels, because temperatures above 500 degrees C [932 degrees F] are required for significant creep in steels [3.9.140]. This indicates that creep of 304L stainless steel is unlikely during the renewal period. Therefore, creep of the assembly hardware is not considered credible in the NAC-UMS System, and aging management is not required during the 60-year timeframe.

3.2.5.2.2 Hydriding

Assembly hardware such as guide tubes and spacer grid materials made from zirconium alloys could potentially be subjected to hydriding effects that could reduce the material's ductility and fracture toughness, particularly at lower temperatures {less than 200 degrees C [392 degrees F]}, once the fuel has cooled [3.9.85].

Hydriding may occur in zirconium alloys that experience hydrogen pickup in reactor service [3.9.20]. As the temperature of the assembly hardware decreases, zirconium hydrides precipitate due to the decreasing hydrogen solubility in the zirconium matrix. The hydride precipitation will occur when the hardware cools in the spent fuel pools after reactor discharge. Some of the hydride will dissolve during the drying process and will reprecipitate due to subsequent cooling during storage. Unlike fuel rods with cladding, there is no hoop stress for the zirconium-based assembly hardware to cause hydride reorientation. Any load on the assembly hardware is predominantly expected due to its own weight, which is not sufficient to be equivalent to hoop stresses to cause hydride reorientation. In addition, any additional hydriding of the assembly hardware during extended storage is expected to be negligible [3.9.94].

In summary, the impact of hydriding effects on assembly hardware, especially guide tubes, is far less severe than for cladding with fuel [3.9.8; 3.9.85]. Because there is limited load during storage on assembly hardware, it is unlikely that hydriding will affect the ability of the assembly hardware to ensure that the spent fuel remains in the as-analyzed configuration. Confirmation of this expectation is provided by Einziger et al. [3.9.12], which did not observe any hydriding effects on assembly hardware after 15 years of dry storage. Therefore, hydriding of assembly hardware components is not considered to be significant, and aging management of the NAC-UMS System is not required during the 60-year timeframe.

3.2.5.2.3 General Corrosion

Various assembly hardware components made of stainless steel or Inconel may be subjected to general corrosion in the presence of humid air or an aqueous solution. General corrosion of assembly hardware made of zirconium alloys is not considered here; it is excluded per the technical basis discussed in Section 3.2.5.1.6. The amount of residual water in the canister during the extended storage is expected to be less than 1 mole per the guidance in NUREG–1536 [3.9.122]. Most residual water is expected to decompose into hydrogen and oxidizing species, such as oxygen and hydrogen peroxide, with time [3.9.94]. However, it is possible for trace amounts of water to remain in the vapor phase in the canister's internal environment for the extended period.

The general corrosion rate of the nickel-based Inconel alloys due to humid air is expected to be on the order of 25 nm/yr [10^{-3} mils/yr] [3.9.359]. The general corrosion rate of 304 stainless steel in the presence of humid air has been reported to be negligible [3.9.313], and the low-carbon grade 304L is expected to behave similarly. Further, as corrosion proceeds, the residual water would deplete with time. Considering the low general corrosion rate of the Inconel alloy, the negligible corrosion rate of 304 stainless steel under humid air conditions, and the radiolysis of the residual water, it is concluded that the effect of general corrosion in the presence of trace amounts of water is insignificant on assembly hardware components during the renewal period. As such, general corrosion of assembly hardware is insignificant, and therefore, aging management of the NAC-UMS System is not required during the 60-year timeframe.

3.2.5.2.4 Stress Corrosion Cracking

Various stainless steel and Inconel assembly hardware components could be susceptible to SCC in the presence of an aggressive environment and sufficient residual tensile stresses. SCC of the structural components may lead to cracking, which can compromise the structural integrity of the component. SCC of assembly hardware made of zirconium alloys is not considered here; it is excluded per the technical basis discussed in Section 3.2.5.1.9.

Residual tensile stresses are expected to be present in the assembly hardware, primarily in welded areas. Regarding the chemical environment, various types of stainless steels are prone to SCC, even in high-purity demineralized water at the temperatures of the BWRs, typically 290 degrees C [554 degrees F] [3.9.318]. This observation is attributed to the presence of dissolved oxygen and other oxidizing species in the primary coolant water [3.9.318] of a BWR. Various types of nickel-based alloys, including Inconel, are susceptible to SCC in the presence of hot water, hot caustic solution, hot wet hydrofluoric acid solution, or aqueous solution containing a sufficient amount of chloride at high temperatures [3.9.344].

In the TSC environment, the water could exist in the liquid state only when the temperature is near or below 100 degrees C [212 degrees F]. The residual water content inside the TSC is expected to be less than 1 mole during dry storage, as per guidance in NUREG–1536 [3.9.122]. During storage, most residual water would decompose into hydrogen and oxidizing species, such as oxygen and hydrogen peroxide, due to radiolysis [3.9.94] . However, it is possible for a trace amount of residual water to persist in the vapor phase of the confinement cavity. The trace amount of water is unlikely to condense into the liquid phase during dry storage because the relative humidity of the DSS internal environment cannot reach 100 percent when the residual amount of water is less than 1 mole. [3.9.94] also evaluated the consequences of much greater levels of residual water within a cask and found that moisture levels up to 17.4 moles [0.313 L]) of water were not sufficient to cause condensation. Further, SCC of stainless steel and Inconel has not been reported in a nonchloride humid air environment.

Because of the lack of halides and the small amount of water in helium and embedded environments, SCC of stainless steel is not considered to be credible. Therefore, aging management of SCC of stainless-steel subcomponents of spent fuel hardware of the NAC-UMS System exposed to helium is not required during the 60-year timeframe.

3.2.5.2.5 Radiation Embrittlement

Radiation embrittlement of assembly hardware such as guide tubes and spacer grid materials made from zirconium alloys is excluded using the basis provided in Section 3.2.5.1.10. Similarly, radiation embrittlement of assembly hardware made of stainless steel or Inconel is not considered credible per the technical bases provided in Sections 3.2.1.9. Therefore, aging management of radiation embrittlement of assembly hardware subcomponents exposed to helium for the NAC-UMS System is not required during the 60-year timeframe.

3.2.5.2.6 Fatigue

Fatigue of assembly hardware such as guide tubes and spacer grid materials made from zirconium alloys is excluded using the basis provided in Section 3.2.5.1.11. Similarly, fatigue of assembly hardware made of stainless steel or Inconel is not considered credible per the technical bases provided in Section 3.2.1.1.7 and 3.2.1.2.7. Therefore, aging management of fatigue of assembly hardware subcomponents exposed to helium for the NAC-UMS System is not required during the 60-year timeframe.

Table 3.2-1 NAC-UMS Transportable Storage Canister (TSC) and Fuel Basket (FB) Aging Management Review (AMR) Results

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-582, Items 1, 2, 3, 4, 5 & 8	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
790-582, Item 6	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
				Radiation Embrittlement	Cracking	No
				Microbiologically Influenced Corrosion	Loss of Material	No
				Fatigue	Cracking	TLAA per Design Code
		Stainless Steel (welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP

Table 3.2-1 NAC-UMS Transportable Storage Canister (TSC) and Fuel Basket (FB) Aging Management Review (AMR) Results

Applicable License Drawing/Item No.	Subcomponent [1]	Material [2]	Storage Operation Environment [3]	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-584, Item 1	Shield Lid	Stainless Steel	FE	Radiation Embrittlement	Cracking	No
				Microbiologically Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	No
			HE	Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	FE	Stress Corrosion Cracking	Cracking	No
790-584, Item 4	Structural Lid	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
790-584, Item 5	Port Cover	Stainless Steel (welded)	FE	Stress Corrosion Cracking	Cracking	No
		Stainless Steel	SH	Radiation Embrittlement	Cracking	No
			FE	Fatigue	Cracking	TLAA per Design Code
				Microbiologically influenced corrosion	Loss of material	No
				Radiation Embrittlement	Cracking	No
				Pitting and Crevice Corrosion	Loss of Material (precursor to SCC)	No

Table 3.2-1 NAC-UMS Transportable Storage Canister (TSC) and Fuel Basket (FB) Aging Management Review (AMR) Results

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-584, Item 7	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
790-584, Item 6	Shield Lid Support Ring	Stainless Steel	HE	Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
790-571, Items 1, 2, and 3, and 790-591, Items 1 - 7	Fuel Basket Bottom Weldments	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 NAC-UMS Transportable Storage Canister (TSC) and Fuel Basket (FB) Aging Management Review (AMR) Results

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-572, Items 1 – 6 and 790-592, Items 1 - 7	Fuel Basket Top Weldments	Stainless Steel	HE	Radiation Embrittlement	Cracking	No
				Fatigue	Cracking	TLAA per Design Code
				Creep	Change in Dimensions	No
		Stainless Steel (welded)	HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
790-573, Item 1 and 790-593, Item 1	Fuel Basket Support Disks	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
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	TLAA
				Creep	Change in Dimensions	No
				Radiation Embrittlement	Cracking	No

Table 3.2-1 NAC-UMS Transportable Storage Canister (TSC) and Fuel Basket (FB) Aging Management Review (AMR) Results

Applicable License Drawing/Item No.	Subcomponent [(1)]	Material [(2)]	Storage Operation Environment [(3)]	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-573, Items 3, 5 – 8; 790-593, Items 2, 3, 5 – 8; 790-570, Item 23; and 790-595, Item 8	Fuel Basket Spacers, Tie Rods, Washers	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
790-573, Item 4 and 790-593, Items 4, 9 and 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
790-574- Item 1 and 790-594, Item 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

Table 3.2-1 NAC-UMS Transportable Storage Canister (TSC) and Fuel Basket (FB) Aging Management Review (AMR) Results

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-575, Items 1, 2, 5, 6 and 7, 790-581, Items 1, 2, 3, 7, 8, 9 and 10, and 790-605, Items 1, 2, 5, 6 and 7	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
790-575, Items 3 and 4; 790-581, Items 4, 5 and 6, and 790-605, Items 3 and 4	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 3.2-1 NAC-UMS Transportable Storage Canister (TSC) and Fuel Basket (FB) Aging Management Review (AMR) Results

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
412-502, Items 2, 5 and 16	Maine Yankee Damaged Fuel Can Lid Plate, Lid Bottom Plate 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
412-502, Items 8, 9, 17 and 18	Maine Yankee Damaged Fuel Can Bottom and 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
412-502, Items 10 and 19	Maine Yankee Damaged Fuel Can 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

Table 3.2-1 NAC-UMS Transportable Storage Canister (TSC) and Fuel Basket (FB) Aging Management Review (AMR) Results

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
412-502, Items 12 and 13	Maine Yankee Damaged Fuel Can 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
412-502, Items 1, 4, 6, 7, 14 and 15	Maine Yankee Damaged Fuel Can Collar, Wiper, and Filter/Backing Screens	Stainless Steel	HE	Creep	Change in Dimensions	No
				Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	No
				Radiation Embrittlement	Cracking	No

Notes

(1) Safety functions and item numbers of TSC and Fuel Basket Subcomponents are identified in Table 2.5-1.
(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 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.
(3) Environments Legend: OD = Air-Outdoor/Air-Indoor; SH = Sheltered; E-C = Embedded in Concrete; FE = Fully Encased (Steel); HE = Helium (Inert Gas).

Table 3.2-2 NAC-UMS Vertical Concrete Cask (VCC) Aging Management Review (AMR) Results

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-561, Items 1, 27, 28, 29, and 30	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	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
790-561, Items 2 and 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
				Pitting and Crevice Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP

Table 3.2-2 NAC-UMS Vertical Concrete Cask (VCC) Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-561, Item 2	Top Flange	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
790-561, Item 16	Weldment Base 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
790-561, Items 10, 11, 13, 14, 15, and 26	Base Weldment (Baffle Weldment) 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
				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

Table 3.2-2 NAC-UMS Vertical Concrete Cask (VCC) Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-561, Item 25	Baffle	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
				Microbiological Influenced Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
790-561, Item 12	Base Weldment Bottom 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
				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
790-561, Item 17	Base Weldment 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 3.2-2 NAC-UMS Vertical Concrete Cask (VCC) Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-561, Items 18, 19, 20, 21, 22, 23, 24 and 26	Outlet Vent Assembly	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
790-561, Items 35 and 790-613, Items 1 and 2	Inlet Vent Supplemental Shield Assembly or 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
790-562, Items 1 – 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 3.2-2 NAC-UMS Vertical Concrete Cask (VCC) Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-562, Item 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
				Creep	Cracking	No
				Dehydration at high temperatures	Cracking	No
					Loss of strength	No
				Delayed ettringite formation	Loss of material (Spalling, Scaling)	No
					Loss of Strength	No
					Cracking	No
				Fatigue	Cracking	No
				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
				Radiation Damage	Cracking	No
					Loss of Strength	No

Table 3.2-2 NAC-UMS Vertical Concrete Cask (VCC) Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-562, Item 15 (continued)	Concrete Shell	Concrete	OD	Shrinkage	Cracking	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
790-562, Items 16, 17, 42, 48, 49 and 50	Inlet/Outlet Screen Assemblies	Stainless Steel	OD	Radiation Embrittlement	Cracking	No
				Pitting and Crevice Corrosion	Loss of Material	No
				Stress Corrosion Cracking	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Stress Relaxation	Loss of Preload	No
790-562, Items 31, 32, and 47	Lifting Anchor Lug, Base Plate and Spacer Plate	Steel	OD	Microbiological Influenced Corrosion	Loss of Material	No
				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
				Radiation Embrittlement	Cracking	No

Table 3.2-2 NAC-UMS Vertical Concrete Cask (VCC) Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-562, Items 31, 32, and 47 (continued)	Lifting Anchor Lug, Base Plate and Spacer Plate	Steel	E-C	General Corrosion	Loss of Material	TLAA
				Pitting and Crevice Corrosion	Loss of Material	TLAA
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
790-562, Items 44 and 45	Lifting Anchor Hardware - Nut and Washer	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
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Stress Relaxation	Loss of Preload	No
790-562, Items 33, 43 and 46	Lifting Anchor Rebar and Threaded Rebar	Steel	E-C	General Corrosion	Loss of Material	TLAA
				Pitting and Crevice Corrosion	Loss of Material	TLAA
				Radiation Embrittlement	Cracking	No
				Microbiological Influenced Corrosion	Loss of Material	No

Table 3.2-2 NAC-UMS Vertical Concrete Cask (VCC) Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-563, Item 1	VCC Lid	Steel	OD	Galvanic corrosion	Loss of Material	TLAA and 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
				Pitting and Crevice Corrosion	Loss of Material	TLAA
				Radiation Embrittlement	Cracking	No
790-564, Items 1, 2, 3, 4, 5, 6, 7, 8, 9, and 10	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

Table 3.2-2 NAC-UMS Vertical Concrete Cask (VCC) Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material (2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-564, Items 1, 2, 3, 4, 5, 6, 7, 8, 9, and 10 (continued)	Shield Plug Assembly	NSC/NSP (NS-3/ NS-4-FR)	FE	Radiation Embrittlement (NS-4-FR only)	Cracking	TLAA
				Thermal Aging	Loss of Fracture Toughness/ Loss of Ductility	TLAA
				Boron Depletion (NS-4-FR only)	Loss of Shielding Effectiveness	TLAA
790-590, Item 13	Lid Bolting Hardware	Stainless Steel	OD	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
790-590, Item 15 (Alternate Assembly Item No. 790-561-36)	Pedestal Cover	Stainless Steel	SH	Stress Corrosion Cracking	Cracking	No
				Radiation Embrittlement	Cracking	No
				Pitting and Crevice Corrosion	Loss of Material	No
				Microbiological Influenced Corrosion	Loss of Material	No

Notes:

(1) Safety functions and item numbers of Concrete Cask Subcomponents are identified in Table 2.5-2.

(2) 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.

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

Table 3.2-3 NAC-UMS Transfer Cask (TFR) and Transfer Adapters Aging Management Review (AMR) Results

Applicable License Drawing/Item No.	Subcomponent (1)	Material2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-560-1	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
790-560, Items 2, 3, 4, 5, and 6	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
790-560, Items 7, 8, 9, 10, and 11	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
790-560-12	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 3.2-3 NAC-UMS Transfer Cask (TFR) and Transfer Adapters Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-560-14	Neutron Shield (Transfer Cask Body and Shield Doors)	NSP (NS-4-FR)	FE	Radiation Embrittlement	Cracking	TLAA
				Thermal Aging	Loss of Fracture Toughness	TLAA
				Boron Depletion	Loss of Shielding Effectiveness	TLAA
790-560-15	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
790-560-16	Shield Door Rails (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	Transfer Cask AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				Wear	Loss of Material	Transfer Cask AMP
790-560-19 & 47 and 790-617-5	Shield Door Lock Bolts / Lock Pins	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 3.2-3 NAC-UMS Transfer Cask (TFR) and Transfer Adapters Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-560-20	Retaining Ring	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
				Galvanic Corrosion	Loss of Material	Transfer Cask AMP
790-560-21	Support Plate	Steel	FE	None Identified	None Identified	No
790-560, Items 26 – 34, and 39	Shield Door Assembly	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
790-560-36	Gamma Shield Brick	Lead	FE	None Identified	None Identified	No
790-560-38	Retaining Ring Bolt	Stainless Steel (Ferritic)	OD	Pitting and Crevice Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Stress Corrosion Cracking	Cracking	No
				Stress Relaxation	Loss of Preload	No
				Microbiological Influenced Corrosion	Loss of Material	No
				Galvanic Corrosion	Loss of Material	Transfer Cask AMP

Table 3.2-3 NAC-UMS Transfer Cask (TFR) and Transfer Adapters Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-560-41	Transfer Cask Extension	Steel	OD	Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
				Microbiological Influenced Corrosion	Loss of Material	No
				General Corrosion	Loss of Material	Transfer Cask AMP
				Radiation Embrittlement	Cracking	No
790-560-42	TFR Extension Bolts	Stainless Steel (Ferritic)	OD	Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	No
				Stress Corrosion Cracking	Cracking	No
				Radiation Embrittlement	Cracking	No
				Stress Relaxation	Loss of Preload	No
790-560-43	Shielding Ring	Steel	FE	None Identified	None Identified	No
790-560-49	Wear Strip	Stainless Steel (Nitronic 30)	OD	Microbiological Influenced Corrosion	Loss of Material	No
				Pitting and Crevice Corrosion	Loss of Material	No
				Radiation Embrittlement	Cracking	No
				Stress Corrosion Cracking	Cracking	No
				Wear	Loss of Material	Transfer Cask AMP

Table 3.2-3 NAC-UMS Transfer Cask (TFR) and Transfer Adapters Aging Management Review (AMR) Results (continued)

Applicable License Drawing/Item No.	Subcomponent (1)	Material 2)	Storage Operation Environment (3)	Aging Mechanism	Aging Effect	Aging Management Activities Required
790-559, Items 1 – 10, 12, 15-18	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
				Microbiological Influenced Corrosion	Loss of Material	No

Notes:

(1) Safety functions and item numbers of NAC-UMS Transfer 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., 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 MTC shield doors and rail components are lubricated with spent fuel pool compatible lubricant such as Neolube or equivalent.

Table 3.2-4 NAC-UMS Spent Fuel Assemblies Aging Management Review (AMR) Results

Structure, System, or Component	Intended Safety Function (1)	Material (2)	Environment (3)	Aging Mechanism	Aging Effect	Aging Management
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 (HBU SNF only)	Cracking	No
				Hydride Reorientation (HBU SNF only)	Cladding Breach/ Structural Failure	High-Burnup Fuel Monitoring and Assessment AMP (HBU SNF only)
				Thermal Creep (HBU SNF only)	Changes in Dimensions	High-Burnup Fuel Monitoring and Assessment AMP (HBU SNF only)
				Low-Temperature Creep (HBU SNF only)	Changes in Dimensions	No
				Radiation Embrittlement	Loss of Strength	No
				Fatigue	Cracking	No
				Mechanical Overload (HBU SNF only)	Cracking	No

Table 3.2-4 NAC-UMS Spent Fuel Assemblies Aging Management Review (AMR) Results

Structure, System, or Component	Intended Safety Function (1)	Material (2)	Storage Environment (3)	Aging Mechanism	Aging Effect	Aging Management
Guide tubes (PWR) or water channels (BWR)	RE, SR	Zirconium-based alloy	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	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 3.2-4 NAC-UMS Spent Fuel Assemblies Aging Management Review (AMR) Results

Structure, System, or Component	Intended Safety Function (1)	Material (2)	Storage Environment (3)	Aging Mechanism	Aging Effect	Aging Management
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 3.2-4 NAC-UMS Spent Fuel Assemblies Aging Management Review (AMR) Results

Structure, System, or Component	Intended Safety Function (1)	Material (2)	Storage Environment (3)	Aging Mechanism	Aging Effect	Aging Management
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 2.5-4.
(2) Materials Legend: CS = 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); GW = Groundwater/Soil.

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-UMS 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 NAC-UMS System 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-UMS 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-UMS System, NAC-UMS System FSARs, 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, 3.2-2, and 3.2-3 for the TSC and Fuel Baskets, VCC, and Transfer Cask.

3.3.3 Evaluation and Disposition of Identified TLAAs

3.3.3.1 Fatigue Evaluation of NAC-UMS System Components for Extended Storage (NAC Calculation No. 30013-2001, Revision 2)

The potential fatigue of the NAC-UMS SSCs (e.g., canisters and fuel baskets) were evaluated in a TLAA for service conditions over the extended period of operation. The NAC-UMS 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-UMS 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-UMS Steel Components for Extended Storage Operation (NAC Calculation No. 30013-2002, Revision 2)

NAC-UMS System Vertical Concrete Cask (VCC) internal sheltered steel SSCs were evaluated for general corrosion at a constant rate of 0.003 inch per year over the entire 60-year service period, for a total corrosion allowance of 0.18 inch. This total corrosion allowance was evaluated and determined not to have an adverse effect on the ability of the VCC's internal assemblies to perform its intended structural, thermal, and shielding functions.

The structural evaluation of the NAC-UMS 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.

The structural evaluation of the NAC-UMS VCC for dead load, live load, flood, tornado wind, and seismic loading did not take any structural credit for the VCC steel inner shell, and therefore, it is concluded that any reduction in the VCC inner shell 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 VCC concrete shell due to tornado missile impacts did not take any structural credit for the VCC steel inner shell, 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 prevent missile perforation.

The structural evaluation of the NAC-UMS VCC assembly for the VCC 24-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 inner shell, 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 1.67-inch, resulting in a 28% reduction of the air inlet cross-section area, which is bounded by the half inlets blocked condition. Furthermore, since the deformation of the corroded pedestal ring is much less than the 6-inch height of the air inlet opening, the weldment plate (and canister) will not “bottom-out”. Due to reduced stiffness of the pedestal, the canister acceleration loads will be lower than those for calculated based on the nominal plate thicknesses.

The structural evaluation of the NAC-UMS 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 NAC-UMS VCC air passages 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 shielding analysis concludes that the reduction in gamma shielding resulting from loss of NAC-UMS VCC steel thickness 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, lifting lug embedments and rebar) 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, 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. 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-UMS Neutron Absorber and Neutron Shield Components (Storage/Transfer) (NAC Calculation No. 30013-5001, Revision 0)

NAC-UMS system was evaluated for:

- Depletion of the neutron absorber Boron-10 content in the basket neutron absorbing materials
  - 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 $1x10^{-9}$ 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).
  - During operation of the NAC-UMS 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-UMS 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 NAC-UMS VCC and transfer cask is well over 60 years.
  - After 60 years < 1% of the B-10 in the neutron shield will be depleted in the most limiting neutron shield component (NAC-UMS transfer cask bottom/door transfer).
- Radiation embrittlement in the cask radiation shield components

- Embrittlement is not a concern for the NAC-UMS system 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 Thermal Aging Evaluation for Type 17-4PH Stainless Steel Support Disks for NAC-UMS PWR Fuel Basket (Appendix A of NAC Calculation No. 30013-2001, Revision 2)

Thermal aging of the 17-4 PH precipitation-hardened stainless steel fuel basket support disks is dispositioned this Section. Based on the discussions in the study by Olender et. al (Ref. 14.6.18), applicable conditions for the 17-4PH support disks of the NAC-UMS PWR basket were evaluated, which include initial heat treatment condition, service temperature, operating environment, and stress level. As shown in the following discussion, thermal aging will not compromise the safety function of the 17-4PH support disks.

Material Condition

The support disks are made of 17-4PH stainless steel per ASME SA-693, Type 630, heat treated to condition H1150. In the recommended actions in the study by Olender et. al, the H1150 is considered optimally heat treated, if the design allows it, which is the case for this application.

Service Temperature

The study by Olender et. al provides guidance for assessing the potential for embrittlement for the 17-4PH stainless steel for operating temperatures between 470°F and 600°F. The maximum temperature of 601°F of the 17-4PH support disks corresponds to the design basis heat load case for the normal condition of storage. This temperature occurs at a very limited region at the center of the middle disk of the basket. The average temperature of all the support disks is 396°F, with temperature at the center of the disks ranging from 294°F to 601°F and the temperature at the disk edges ranging from 165°F to 400°F. Note that this temperature profile corresponds to the design basis heat load condition. Also note that the operating temperatures for all the system components continually decrease during the lifetime of the storage as a result of fuel decay.

Operating Environment

Most cases of the failures of 17-4PH components observed and described in the study by Olender et. al occurred in active components such as valves or subcomponents of valves, which were in a pressure boundary and were subjected to dynamic loading. This indicates that the failures are associated with large primary stresses. Additionally, much of the operating experiences described in the study indicate an element of corrosion, SCC or overloading contributing to the observed failure.

During the long-term storage condition, the basket support disks maintain the locations of each fuel tube for criticality control. The support disks are individually supported at eight tie-rod locations and are subjected to static load from the self-weight of the disks only. The support disks are not subjected to dynamic or cyclic loads.

Since the canisters are backfilled with helium, the support disks are in an inert environment, not an aggressive chemical environment as discussed in the study by Olender at. el. Also, much of this operating experience in the study is in a system pressure boundary or subject to dynamic loads on an active component. The loading condition and the operating environment for the support disks in the long-term storage conditions are static and mild. This type of operating environment would not challenge the safety function of the support disks exposed to any potential thermal aging effects.

Stress Level

During the long-term storage condition, the support disks are in a horizontal position and individually supported at eight tie-rod locations. Each support disk is subjected to a static inertial load corresponding to its self-weight only, resulting in minimal stresses in the disk. As shown in Table 3.4.4.1-12 of the NAC-UMS FSAR, the maximum stress intensity for the primary membrane plus primary bending (P_m + P_b) stress is 0.8 ksi. Using the allowable stress as defined by the ASME Code Subsection NG is 52.7 ksi, which is based on a conservative temperature of 800°F, the minimum margin of safety for the support disks is 64.8. Since the support disks are under such a low stress level, any thermal aging effect on this favorably heat-treated material would have a negligible effect on the structural function of the support disk.

Conclusion of Evaluation

The 17-4PH support disks for NAC-UMS PWR system are in a service where some portion of them can be exposed to operating temperatures between 470°F and 600°F, which can potentially cause embrittlement. However, the component failures described in the study by Olender at. el. are associated with the combined conditions of high stresses due to dynamic or cyclic loads and elevated operating temperatures. Therefore, based on the discussion above, any thermal aging effect will not adversely impact the safety function of the 17-4PH support disks because of the following characteristics and conditions.

- The support disks have been heat treated to an optimum condition to minimize the susceptibility of the material to thermal embrittlement.
- Only a limited portion of the disks are subjected to temperatures above 470°F and the average disk temperature is below 400°F for normal conditions of storage (Note that all disks are subjected to negligible static primary stresses).
- The disks are stainless steel materials in an inert operating environment. The loading condition and the operating environment for the support disks in the long-term storage conditions are static and mild.
- The support disks are subjected to insignificant static loads (self-weight only) with a very low stress level.

3.4 AGING MANAGEMENT PROGRAMS (AMPs)

This section lists and describes the proposed AMPs identified as required in Section 3.2 for the NAC-UMS 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 that are subject to aging effects that require either an AMP or TLAA are identified in Section 3.2 and 3.3. 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 period of operation. 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, 3.2-2, 3.2-3 and 3.2-4 for the TSC and Fuel Baskets, VCC, Transfer Cask and Transfer Adapter, and SNF Assemblies, respectively. 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-UMS Technical Specifications, with additional examinations to address aging that could potentially occur during the period of extended operation.

The identified AMPs and a discussion of the bases for the inspections/examinations specified in the AMPs are detailed as follows:

- Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSCs) (Table A-1).
- Aging Management Program Internal VCC Metallic Components Monitoring (Table A-2).
- Aging Management Program for External VCC Metallic Components Monitoring (Table A-3)
- Aging Management Program for Reinforced Vertical Concrete Cask (VCC) Structures Concrete Monitoring (Table A-4)
- Aging Management Program for Transfer Casks (TFRs) and Transfer Adapters (Table A-5)
- Aging Management Program for NAC-UMS High Burnup Fuel Monitoring and Assessment (Table A-6)

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 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-UMS 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 (3.9.370) for supplemental examination of major indications as documented in the proposed AMP. To support identification of the most susceptible TSCs to SCC, all NAC-UMS 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 (3.9.370). 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-UMS Steel Components for Extended Storage Operation (NAC Calculation No. 30013-2002) 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-UMS Steel Components for Extended Storage Operation (NAC Calculation No. 30013-2002) 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 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 [3.9.154] 3 tier concrete evaluation criteria. VCC fully embedded components including Nelson studs, rebar, and lifting lug embedments were evaluated by TLAA Corrosion Analysis of NAC-UMS Steel Components for Extended Storage Operation (NAC Calculation No. 30013-2002) 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-UMS System shielding performance [3.9.369], it has been shown that cracks that are less than 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 is noted that all NAC-UMS 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.

3.4.3.6 AMP-6 Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment

The High-Burnup (HBU) Fuel Aging Management Program is appliable to all General Licensees (GLs) storing uncanned HBU fuel assemblies (≥ 45 GWd/MTU) in the NAC-UMS storage system. HBU fuel stored in the NAC-UMS System is limited to a maximum assembly average burnup of 60 GWd/MTU. The AMP relies on the joint EPRI and DOE HBU Dry Storage Cask Research and Development Project (HDRP) as a surrogate demonstration program that monitors the performance of HBU fuel in dry storage. After fuel loading operations are completed, the loaded TSC is drained and vacuum dried in accordance with CoC No. 1015 Technical Specification (TS) A3.1.2 followed by high purity helium backfill to atmospheric pressure in accordance with TS A3.1.3. These two TSs ensure that the HBU fuel is stored in a dry, inert environment, thus preventing cladding degradation due to oxidation mechanisms. The NAC-UMS TSCs are loaded and processed in accordance with ISG-11, Revision 3. The AMP identifies the fuel and canister parameters to be monitored and establishes a detailed list of acceptance criteria to confirm during the surrogate monitoring program. The AMP also identified the corrective actions to be taken by GLs if any of the identified acceptance program are not met. The proposed HBU Fuel Monitoring Program AMP is in compliance with Table 6-7 example AMP provided in NUREG – 2214.

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 state of knowledge of aging-related operational experience, research, monitoring, and inspections to ascertain the ability of in-scope NAC-UMS 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 inservice 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-UMS 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, 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-UMS 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-UMS 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-UMS 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-UMS canister to the approved and NRC certified NAC-UMS transport cask system (CoC No. 71-9270) [3.9.155] 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. PWR SNF assemblies stored in the NAC-UMS storage system include low burnup (≤ 45 GWd/MTU), intact and damaged loaded in DFCs, and a limited number of high burnup (≥ 45 GWd/MTU), zircaloy alloy clad loaded in DFCs, and high burnup fuel stored without secondary canning in DFCs. Uncanned high burnup fuel will be evaluated based on the High Burnup Fuel Monitoring and Assessment AMP which relies on the periodic results of the EPRI and DOE HBU Dry Storage Cask Research and Development Project. Degradation of the cladding of low burnup and high 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-UMS canister to be lifted and transferred to a NAC-UMS transport cask. During the AMR, the appropriate NAC-UMS 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-UMS 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-UMS System operating data has been completed and documented to identify any off-normal, accident, or other event potentially effecting the performance of the NAC-UMS System. Based on the review performed on data submitted by the three NAC-UMS System GLs, no significant normal operating event has been identified that would have a significant effect on overall system performance. 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 and Conclusions

3.7.2.1 Executive Summary

During the week of July 23 thru July 26, 2018, the Pre-Application Inspection of NAC-UMS System No. Vertical Concrete Cask (VCC) #55 and Transportable Storage Canister (TSC) #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. A non-proprietary copy of the final approved inspection report is provided in Appendix E.

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 #55 / TSC #22 was selected for the pre-application inspection in accordance with the criteria of EPRI Technical Report, TR-3002005371 [3.15] as documented in NAC Technical Report No. ED20170046, "NAC-UMS and NAC-MPC ISFSI". TSC rankings based on EPRI CISCC Criteria, dated April 18, 2017 [3.16]. The assessment included an analysis to determine the bounding NAC-UMS System or NAC-MPC System from the combined system fleets of 302 deployed 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),

3.7.2.2 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.

3.7.2.3 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:

<u>Transportable Storage Canister (TSC):</u>

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.

VCC Internal Metallic Inspection:

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.

<u>External VCC Metallic Components Inspection:</u>

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 review of all NAC-UMS System design bases documents has been performed to support the TLAA and AMP processes. A complete database of applicable NAC-UMS 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-UMS System SSCs.

None of the design basis documents reviewed affirmatively met the six questions identified in NUREG-1937 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

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Appendix A

Aging Management Program

NAC-UMS (CoC 72-1015)

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LIST OF TABLES

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	<u>Method or Technique</u> 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. <u>Sample Size</u> 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.

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)	<u>Frequency</u> • Baseline inspection at beginning of the period of extended operation • Subsequently every 10 years for TSCs without detection of indications of major corrosion degradation or SCC • Subsequently every 5 years for TSCs with detection of major indications of corrosion degradation or detection(s) of SCC <u>Data Collection</u> 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. <u>Timing of Inspections</u> The baseline inspection shall be performed within 1-year after the 20^{th} anniversary of the first cask loaded at the ISFSI, or within 1-year after the effective date of the CoC renewal if the 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

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	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) 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 (2-inch) 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.

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)	6.4 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
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-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
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 TSC Systems</u> 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 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	<u>Method or Technique</u> 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.

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

AMP Element	AMP Description
4. Detection of Aging Effects (continued)	<u>Sample Size</u> These are opportunist inspections conducted in conjunction with TSC inspections. This inspection is performed when the TSC inspection is conducted. <u>Frequency</u> These are opportunist inspections conducted in conjunction with TSC inspections. This inspection is performed when the TSC inspection is conducted. <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> These are opportunist inspections conducted in conjunction with TSC inspections. This inspection is performed when the TSC inspection is conducted.
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 the IWE-3511 standard or their equivalent.

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

AMP Element	AMP Description
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. One VCC lid bolt was replaced following lid removal due to corrosion. • 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 the subject inspection report in Appendix E. 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, 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. <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 20^{th} 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 or lift lug hardware and loose bolts on VCC lid and lifting lug bolting, 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. <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 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 structures 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. • Increase in Gamma dose rates exceeding LCO A 3.2.2 levels.
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, rebar corrosion, 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 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 new tier 2 findings may be accepted after review by the designated responsible-in-charge engineer. • All new tier 3 findings must be reviewed by the designated responsible-in-charge engineer and are subject to further evaluations as appropriate for the finding. New tier 3 indications or any other indication which could potentially increase dose rates shall be inspected with Gamma Dose rate measurements and verified to be less than LCO A 3.2.2 acceptance criteria 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) • Increases in Gamma Dose rates exceeding LCO A 3.2.2 acceptance criteria
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	<u>Method or Technique</u> 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 or their equivalent. • Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent.

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)	<u>Sample Size</u> All NAC Transfer Casks/Transfer Adapters. <u>Frequency</u> 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. <u>Data Collection</u> Documentation of the inspections required by this AMP, shall be added to 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).

Table A-5
AMP-5 - Aging Management Program for Transfer Casks (TFRs) and Transfer Adapters (continued)

AMP Element	AMP Description
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.
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.

Table A-6
AMP-6 - Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment

AMP Element	AMP Description
1. Scope of Program	The High-Burnup (HBU) Fuel Aging Management Program (AMP) is a generic program applicable to all licensees storing uncanned HBU (≥ 45 GWd/MTU) Fuel Assemblies (FAs) in the NAC-UMS System [(1)] beginning in March 2009. HBU FAs stored in the NAC-UMS System are limited by Technical Specification to an assembly average burnup of ≤ 60 GWd/MTU. The maximum HBU FA burnup loaded into a NAC-UMS System had a nominal burnup of < 55 MWd/MTU and was loaded on July 18, 2014. The HBU FAs have Zirconium alloy fuel cladding (e.g., Zirc-4, ZIRLO, low-tin Zirc-4, and M5). All HBU FAs will be stored in a high-purity Helium atmosphere and are bounded by the by the temperature limits of ISG-11, Revision 3. The program is to manage the factors that could affect the ability to comply with 10 CFR 72.122(h)(1) including fuel cladding temperature, fuel cladding breach, assembly distortion, residual moisture after drying, changes in hydride structure of the cladding, and cladding creep.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions. During the initial loading operations of the NAC-UMS Transportable Storage Canisters (TSCs) the design bases and Certificate of Compliance (CoC) and Technical Specifications (TS) require that the fuel be stored in a dry inert environment. TS Limiting Condition of Operation (LCO) A3.1.2 “Canister Vacuum Drying Pressure” demonstrates that the TSC cavity is dry by maintaining a cavity absolute pressure less than or equal to 10 torr for 10 minutes with the TSC isolated from the vacuum pump. Following the dryness verification, the TSCs are then re-evacuated to ≤ 3 torr, backfilled with 99.9% pure helium to atmospheric, then re-evacuated to ≤ 3 torr for final helium backfill in accordance with TS LCO A3.1.3 “Canister Helium Backfill Pressure”. These two TS requirements ensure that the HBU fuel is stored in a dry and inert environment, thus preventing cladding degradation due to oxidation mechanisms. TS LCO A3.1.1 “Canister Maximum Time in Vacuum Drying” prescribes times that the helium environment be established after commencing TSC draining for design bases heat loads. HBU fuel contents loaded with total heat load of less than 17.4 kW had 33 hours to establish a final helium backfill. The NAC-UMS TSCs are loaded and processed in compliance within the temperature limits of ISG-11, Revision 3.

[(1)] HBU fuel assemblies (maximum burnup < 50 GWd/MTU) were loaded at Maine Yankee Atomic Power Plant (MY) in specially designed NAC-UMS damaged fuel cans (DFCs). A maximum of four MY HBU fuel assemblies in DFCs were preferentially loaded in four corner locations of the 24 PWR fuel basket in accordance with CoC Technical Specifications, Appendix B requirements. Therefore, this AMP is not applicable to MY canned HBU fuel in NAC-UMS Systems.

Table A-6
AMP-6 - Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment (continued)

AMP Element	AMP Description
3. Parameters Monitored/ Inspected	The surrogate demonstration program, e.g., DOE/EPRI's HDRP, parameters monitored and/or inspected will include: • Fuel cladding temperature • Cavity gas temperature, pressure, and composition The monitoring of the above parameters supplemented by physical examination of should be able to provide confirmation if: • The models of degradation phenomena used for 20-year predictions can be used for the TLAA beyond 20 years. • The condition of the fuel, after an appropriately long period of storage, does not degrade. • New degradation mechanisms are not being exhibited.
4. Detection of Aging Effects	Since limited AMP action can be taken inside a sealed TSC, this program relies on the surrogate monitoring inspections of the DOE's HBU Dry Storage Cask Research and Development Project (HDRP) to verify no unexpected aging effects or to identify aging effects if they occur.
5. Monitoring and Trending	Monitoring and trending methods will be used to: • Assess information/data from the HDRP or from other sources (such as testing or research results and scientific analyses) when it becomes available. • The licensees will monitor, evaluate, and trend the information via their operating experience program and/or CAP to determine what actions should be taken to manage fuel and cladding performance, if any. • Formal evaluations (Tollgates) of the aggregate information from the HDRP and other available domestic or international operating experience (including data from monitoring and inspection programs, NRC-generated communications, and other information) will be performed at specific points in time during the period of extended operation, as delineated in Table 14.5-2 of the NAC-UMS FSAR. • If any of the acceptance criteria of Element 6 are not met, the licensee must conduct additional assessments and implement appropriate corrective actions per Element 7.

Table A-6
AMP-6 - Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment (continued)

AMP Element	AMP Description
6. Acceptance Criteria	If any of the following fuel performance criteria are not met in the HDRP, a corrective action is required: • Cladding Temperature – The maximum cladding temperature measured is less than or equal to that predicted by the thermal analysis methods in the UFSAR for the affected TSCs • Cavity Gas Temperature – The average cavity gas temperature measured is less than equal to the that predicted by the thermal analysis methods in the FSAR for the affected TSCs • Cavity Gas Pressure – The cavity gas pressure measured should correspond to the gas pressure predicted for the thermal conditions and corresponding to < 1.0% failed fuel rods. • Cladding Creep – Total creep strain extrapolated to the to the total storage duration based on the best fit to the data accounting for initial condition uncertainty shall be less than 2.5% • Hydrogen content – Maximum hydrogen content of the cover gas over the approved storage period should be extrapolated from the gas measurements to be less than the design-bases limit for hydrogen content for the NAC-UMS Storage System of < 4% of free volume. • Drying –The moisture content in the TSC, accounting for measurement uncertainty, should be less than the expected upper-bound moisture content of < 0.5 gm-mole. • Fuel condition/performance–nondestructive examination (e.g., fission gas analysis) and destructive examination (e.g., to obtain data on creep, fission gas release, hydride reorientation, cladding oxidation, and cladding mechanical properties) confirms the design-bases fuel condition of ≤ 1% rod failure per the analyzed fuel configuration considered in the NAC-UMS System FSAR and approved design bases. • Fuel Rod Breach – Fission gas analysis shall not indicate more than 1.0% of fuel rod cladding breaches.

Table A-6
AMP-6 - Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment (continued)

AMP Element	AMP Description
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). Corrective actions should be implemented if data from the HDRP or other sources of information indicate that any of the HDRP acceptance criteria are not met. If any of the acceptance criteria are not met, the licensee will: • Assess fuel performance (impacts on fuel and changes to fuel configuration), including any consequences of above-design-basis moisture levels on potential degradation of the fuel assembly. • Assess the design-bases safety analyses, considering degraded fuel performance (and any changes to fuel configuration), to determine the ability of the NAC-UMS System to continue to perform its intended functions under normal, off-normal, and accident conditions. The licensee will determine what corrective actions should be taken to: • Manage fuel performance, if any • Manage impacts related to degraded fuel performance to ensure that all intended functions for the NAC-UMS are met In addition, the licensee will obtain the necessary NRC approval in the appropriate licensing/certification process for modification of the design bases to address any conditions outside of the approved design bases.
8. Confirmation Process	The confirmation processes will be commensurate with the general 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 describes or references procedures to: • Determine follow-up actions to verify effective implementation of corrective actions • monitor for adverse trends due to recurring or repetitive finding or observations.

Table A-6
AMP-6 - Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment (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. The HBU Fuel Tollgate program described in FSAR Chapter 14, Table 14.5-2 will reference and evaluate applicable operating experience on a best-effort basis, including: • Internal and industrywide condition reports. • Internal and industrywide corrective action reports. • Vendor-issued safety bulletins. • NRC Information Notices and NUREGs • IAEA Publications on OE for HBU Fuel Performance • Applicable DOE or industry initiatives (e.g., HDRP). • Applicable research (e.g., Oak Ridge National Laboratory studies on bending responses of the fuel, Argonne National Laboratory and Central Research Institute of Electric Power Industry studies on hydride reorientation effects). The review of OE clearly identifies any HBU fuel degradation as either age related, or event driven, with proper justification for that assessment. Past operating experience supports the adequacy of the HDRP. NAC-UMS general licensees have no specific OE on the performance of HBU fuel, in dry storage to date.

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Appendix B

Time-Limited Aging Analysis

NAC-UMS (CoC 72-1015)

TABLE OF CONTENTS

B1.0 INTRODUCTION

The NAC-UMS systems are utilized for spent fuel storage at facilities in the United States. The Nuclear Regulatory Commission initially issued a 20-year 10 CFR Part 72 CoC (72-1015) for this system on November 20, 2000. The license renewal application is required to contain an evaluation of Time-Limited Aging Analysis (TLAA) to demonstrate the safe operation over the extended service life for the cask system. The TLAAs prepared for this renewal application are contained in this Appendix and are comprised of the following items:

- Fatigue Evaluation of MPC and UMS Storage System Components for Extended Storage, 30013-2001, Revision 2
- Corrosion Analysis of UMS VCC Steel Components for Extended Storage, 30013-2002, Revision 2
- Aging Analyses for MPC/UMS Neutron Absorber and Neutron Shield Components (Storage/Transfer), 30013-5001, Revision 0

CALCULATIONS ARE PROPRIETARY AND WITHHELD
IN THEIR ENTIRETY PER 10 CFR 2.390

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Appendix C

Final Safety Analysis Report Changed Pages

NAC-UMS (CoC 72-1015)

TABLE OF CONTENTS

C1.0 INTRODUCTION

This appendix provides a supplement and identifies pertinent changes to the NAC-UMS 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-UMS ITS systems, structures, and components (SSCs). This proposed new UFSAR Chapter will also present the results of the evaluations of time-limited aging analyses (TLAAs) for the renewed license period.

C2.0 CHANGES TO EXISTING UFSAR INFORMATION

List of Changes for the NAC-UMS FSAR, Revision 22A

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	
Page 1.2-2	Modified first and second paragraph where indicated.
Page 1.2-3 thru 1.2-10	Text flow 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 9.2-2	Revised paragraph at the end of section 9.2.1
Page 9.2-3	Revised paragraph at the end of section 9.2.2
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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1.2 General Description of the Universal Storage System

The Universal Storage System provides long-term storage of any of three classes of PWR fuel or two classes of BWR fuel, and subsequent transport using a Universal Transport Cask (Docket 71-9270). During long-term storage, the system provides 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.

1.2.1 Universal Storage System Components

The design and operation of the principal components of the Universal Storage System and the associated ancillary equipment are described in the following sections. The weights of the principal components are provided in Section 3.2.

The Universal Storage System consists of three principal components:

- Transportable Storage Canister (including PWR or BWR fuel basket),
- Vertical Concrete Cask, and
- Transfer Cask/Transfer Adapter.

The design characteristics of these components are presented in Table 1.2-1.

Ancillary equipment needed to use the Universal Storage System are:

- Automated or manual welding equipment;
- An air pallet or hydraulic roller skid (used to move the concrete cask on and off the heavy haul trailer and to position the concrete cask on the storage pad);
- Suction pump, vacuum drying, pressure testing, helium backfill and leak detection equipment;
- A heavy haul trailer or transporter (for transport of concrete cask to the storage pad); or a Vertical Cask Transporter to lift and move the concrete cask in place of air pallets/heavy haul trailer;
- An adapter plate and hardware to position the transfer cask with respect to the storage or transport cask; and
- A lifting yoke for the transfer cask and lifting slings for the canister and canister lids.

In addition to these items, the system requires utility services (electric, helium, air and water), common tools and fittings, and miscellaneous hardware.

For decommissioned sites that have disposed of their auxiliary equipment, new or refurbished equipment shall be used to allow the successful transfer of the NAC-UMS Transportable Storage Canisters (TSCs) from the VCC to the appropriate NAC-UMS System transport cask certified for the transport of the NAC-UMS TSCs. New transfer equipment including the Transfer Cask and Transfer Adapter shall be procured, inspected, and tested in accordance with approved NAC Procurement Specification and applicable License Drawings. If currently existing equipment is to be utilized, the refurbished equipment will be required to comply with all requirements of the applicable Transfer Cask and Transfer Adapter Aging Management Program, as will equipment that is in storage at some of the decommissioned sites prior to delivery to the NAC-UMS Licensees.

Other auxiliary equipment not addressed by the NAC-UMS aging management program (e.g., lifting yokes/extensions, transport equipment, vacuum and leak test systems, required welding equipment, etc.) will be procured by the Licensee or provided by the transport cask contractor.

1.2.1.1 Transportable Storage Canister

Three classes of Transportable Storage Canisters accommodate the PWR fuel assemblies, and two classes of Transportable Storage Canisters accommodate the BWR fuel assemblies. The canister is designed to be transported in the Universal Transport Cask. Transport conditions establish the design basis load conditions for the canister, except for canister lifting. The transport load conditions produce higher stresses in the canister than would be produced by the storage load conditions. Consequently, the canister design is conservative with respect to storage conditions. The evaluation of the canister for transport conditions is documented in the Safety Analysis Report for the Universal Transport Cask, Docket No. 71-9270.

The Transportable Storage Canister consists of a stainless steel canister that contains the fuel basket structure and contents. The canister is defined as confinement for the spent fuel during storage and is provided with a double welded closure system. The welded closure system prevents the release of contents in any design basis normal, off-normal or accident condition. The basket assembly in the canister provides the structural support and primary heat transfer path for the fuel assemblies while maintaining a subcritical configuration for all normal conditions of storage, off-normal events and hypothetical accident conditions. The PWR and BWR fuel basket assemblies are discussed in Section 1.2.1.2.

The major components of the Transportable Storage Canister are the shell and bottom, basket assembly, shield lid, and structural lid. The canister and the shield and structural lids provide a confinement boundary during storage, shielding, and lifting capability for the basket. The

Transportable Storage Canister design parameters for the storage of the five classes of fuel are provided in Table 1.2-2.

The canister consists of a cylindrical, 5/8-inch thick Type 304L stainless steel shell with a 1.75-inch thick Type 304L stainless steel bottom plate and a Type 304 stainless steel shield lid support ring. A basket assembly is placed inside the canister. The shield lid assembly is a 7-inch thick Type 304 stainless steel disk that is positioned on the shield lid support ring above the basket assembly. The shield lid is welded to the canister after the canister is loaded and moved to the workstation for completion of canister closure activities. Two penetrations through the shield lid are provided for draining, vacuum drying, and backfilling the canister with helium. The drain pipe is threaded into the shield lid after the canister is moved to the workstation. The vent penetration in the shield lid is used to aid water removal and for vacuum drying and backfilling the canister with helium. After the shield lid is welded in place, it is pressure and leakage tested to ensure no credible leakage of the confinement boundary during storage.

The structural lid is a 3-inch thick Type 304L stainless steel disk positioned on top of the shield lid and welded to the shell after the shield lid is welded in place and the canister is drained, dried, and backfilled with helium. Removable lifting fixtures, installed in the structural lid, are used to lift and lower the loaded canister.

The Transportable Storage Canister is designed to the requirements of the ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Division I, Subsection NB [8]. It is fabricated and assembled in accordance with the requirements of Subsection NB to the maximum extent practicable, consistent with the conditions of use. Exceptions to the ASME Code are noted in Table B3-1 in Appendix B of the CoC.

A summary of the canister fabrication specifications is presented in Table 1.2-3. As shown in that table, the field installed welds joining the shield and structural lids to the canister shell are not full penetration welds. The shield lid weld is dye penetrant inspected on the root and final cover pass. The structural lid weld is either ultrasonically inspected when completed or it is dye penetrant inspected on the root and final cover passes and on each 3/8-inch intermediate layer. These inspections assure weld integrity in accordance with the requirements of ASME Code Section V, Articles 5 and 6 [9], as appropriate. The weld joining the shield lid to the canister shell is pressure tested and leak tested as described in Section 8.1.1. The structural and shield lid welds are made with the aid of a backing ring (also called a spacer ring) or shims, which cannot be removed when the weld is completed. There are no detrimental effects that result from the presence of the spacer ring or shims, and no structural credit is taken for their presence.

The design of the transportable storage canister and its fabrication controls would allow the canister to be ASME Code stamped in accordance with the ASME Code Section III, if desired.

1.2.1.2 Fuel Baskets

The transportable storage canister contains a fuel basket which positions and supports the stored fuel in normal, off-normal and accident conditions. As described in the following sections, the design of the basket is similar for the PWR and BWR configurations. The fuel basket for each fuel type is designed and fabricated to the requirements of the ASME Code, Section III, Division I, Subsection NG [10]. However, the basket assembly is not Code stamped and no reports relative to Code stamping are prepared. Consequently, an exception is taken to Article NG-8000, Nameplates, Stamping and Reports.

1.2.1.2.1 PWR Fuel Basket

The PWR fuel basket is contained within the transportable storage canister. It is constructed of stainless steel, but incorporates aluminum disks for enhanced heat transfer. The fuel basket design is a right-circular cylinder configuration with square fuel tubes laterally supported by a series of support disks. The basket design parameters for the storage of the three classes of PWR fuel are provided in Table 1.2-4. The Class 1, 2 or 3 fuel baskets incorporate 30, 32 or 34 support disks, respectively. The disks are retained by a top nut and supported by spacers on tie rods at eight locations. The top nut is torqued at installation to provide a solid load path in compression between the support disks. The support disks are fabricated of SA-693, Type 630, 17-4 PH stainless steel. The disks are spaced axially at 4.92 inches center-to-center and contain square holes for the fuel tubes.

The top and bottom weldments are fabricated from Type 304 stainless steel and are geometrically similar to the support disks. The tie rods and top nuts are fabricated from SA-479, Type 304 stainless steel. The top nut is fabricated from a 3.5-in.-diameter bar, and the spacers are fabricated from a 2.5-in. pipe XXS, Type 304 stainless steel. The fuel tubes are fabricated from A-240, Type 304 stainless steel and support an enclosed neutron absorber sheet on each of the four sides. The neutron absorber provides criticality control in the basket. No credit is taken for the fuel tubes for structural strength of the basket or support of the fuel assemblies.

Each PWR fuel basket has a capacity of 24 PWR fuel assemblies in an aligned configuration in 8.80-inch square fuel tubes. The holes in the top weldment are 8.75-inch square. The holes in the bottom weldment are 8.65-inch square. The basket design traps the fuel tube between the top and bottom weldments, thereby preventing axial movement of the fuel tube. The support disk configuration includes webs between the fuel tubes with variable widths depending on location.

The PWR basket design incorporates Type 6061-T651 aluminum alloy heat transfer disks to enhance heat transfer in the basket. Twenty-nine heat transfer disks are contained in the Class 1 basket. Class 2 and 3 fuel baskets contain 31 and 33 disks, respectively. The heat transfer disks are spaced and supported by the tie rods and spacers, which also support and locate the support disks. The heat transfer disks, located at the center of the axial spacing between the support disks, are sized to eliminate contact with the canister inner shell due to differential thermal expansion.

The Transportable Storage Canister is designed to facilitate filling with water and subsequent draining. Water fills and drains freely between the basket disks through three separate paths. One path is the gaps that exist between the disks and canister shell. The second path is through the gaps between the fuel tubes and disk that surrounds the fuel tubes. The third path is through three 1.3-inch diameter holes in each of the disks that are intended to provide additional paths for water flow between disks. The basket bottom weldment supports the fuel tubes above the canister bottom plate. The fuel tubes are open at the top and bottom ends, allowing the free flow of water from the bottom of the fuel tube. The bottom weldment is positioned by supports 1.0 inch above the canister bottom to facilitate water flow to the drain line. These design features ensure that water flows freely in the basket so that the canister fills and drains evenly.

1.2.1.2.2 BWR Fuel Basket

Like the PWR fuel basket, the BWR basket is contained within the stainless steel Transportable Storage Canister. The BWR fuel basket is also a right-circular cylinder configuration with square fuel tubes laterally supported by a series of support disks (40 disks for the Class 4 fuel basket and 41 disks for the Class 5 fuel basket). The basket design parameters for the storage of the two classes of BWR fuel are provided in Table 1.2-4. The support disks are retained by cylindrical spacers on tie rods at six locations. The top nut is torqued at installation to provide a solid load path in compression between the support disks. The support disks are fabricated of SA-533, Type B, Class 2 carbon steel and are coated with electroless nickel to inhibit corrosion and the formation of combustible gases during fuel loading. The disks are spaced axially at 3.8-inch center-to-center and contain square holes for the fuel tubes.

The top and bottom weldments are fabricated from Type 304 stainless steel, and are geometrically similar to the support disks. The fuel tubes are also fabricated from Type 304 stainless steel. Three types of tubes are designed to contain one BWR fuel assembly: tubes with neutron absorber on two sides, tubes with neutron absorber one side, and tubes with no neutron absorber. No credit is taken for the fuel tubes for structural strength of the basket or support of the fuel assemblies.

Each BWR fuel basket has a capacity of 56 BWR fuel assemblies in an aligned configuration. The fuel tubes in 52 positions have an inside square dimension of 5.90 inches. The inside dimension of the four fuel tubes located in the outside corners of the basket array is 6.05-inches square. The holes in the top weldment are 5.75 inches by 5.75 inches, except for the four enlarged holes, which are 5.90 inches-square. The holes in the bottom weldment are 5.63-inches square. The basket design traps the fuel tube between the top and bottom weldments, thereby preventing axial movement of the fuel tube. The support disk webs between the fuel tubes are 0.65-inch wide. The BWR fuel basket design also incorporates 17 Type 6061-T651 aluminum alloy heat transfer disks similar in design and function of those in the PWR baskets.

The BWR canister is also designed to facilitate filling with water and subsequent draining. Water fills and drains freely between the basket disks through three separate paths. One path is the gaps that exist between the disks and canister shell. The second path is through the gaps between the fuel tubes and disk that surrounds the fuel tubes. The third path is through three 1.3-inch diameter holes in each of the disks that are intended to provide additional paths for water flow between disks. The basket bottom weldment supports the fuel tubes above the canister bottom plate. The fuel tubes are open at the top and bottom ends, allowing the free flow of water from the bottom of the fuel tube. The bottom weldment is positioned by supports 1.0 inch above the canister bottom to facilitate water flow to the drain line. These design features ensure that water flows freely in the basket so that the canister fills and drains evenly.

1.2.1.3 Vertical Concrete Cask

The Vertical Concrete Cask is the storage overpack for the Transportable Storage Canister. Five concrete casks of different lengths are designed to store five canisters of different lengths containing one of three classes of PWR or of two classes of BWR fuel assemblies. The concrete cask provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the canister during long-term storage. Table 1.2-5 lists the principal physical design parameters of the concrete cask.

The concrete cask is a reinforced concrete (Type II Portland cement) structure with a structural steel inner liner. The concrete wall and steel liner provide the neutron and gamma radiation shielding to reduce the average contact dose rate to less than 50 millirem per hour for design basis PWR or BWR fuel. 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 concrete cask incorporates reinforced chamfered corners at the edges to facilitate construction. The concrete cask is shown in Figure 1.2-1.

The Vertical Concrete Cask forms 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 to the concrete cask cavity to minimize radiation streaming. A baffle assembly directs inlet air upward and around the pedestal that supports the canister. The weldment structure includes the baffle assembly configuration, as shown in Drawing 790-561. The decay heat is transferred from the fuel assemblies to the tubes 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 of the 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 top of the Vertical Concrete Cask is closed by a shield plug and lid. The shield plug is approximately 5 inches thick and incorporates carbon steel plate as gamma radiation shielding, and NS-4-FR or NS-3 as neutron radiation shielding. A carbon steel lid that provides additional gamma radiation shielding is installed and bolted in place above the shield plug. The shield plug and lid reduce skyshine radiation and provide a cover and seal to protect the canister from the environment and postulated tornado missiles. At the option of the user, a tamper-indicating seal wire and seal may be installed on two of the concrete cask lid bolts. An optional supplemental shielding fixture, shown in Drawing 790-613, may be installed in the air inlets to reduce the radiation dose rate at the base of the cask.

Fabrication of the concrete cask involves no unique or unusual forming, concrete placement, or reinforcement requirements. The concrete portion of the concrete cask is constructed by placing concrete between a reusable, exterior form and the inner metal liner. Reinforcing bars are used near the inner and outer concrete surfaces, to provide structural integrity. The inner liner and base of the concrete cask are shop fabricated. The principal fabrication specifications for the concrete cask are shown in Table 1.2-6.

1.2.1.4 Transfer Cask

The transfer cask is a heavy lifting device, which is designed, fabricated, and load-tested to meet the requirements of NUREG-0612 [11] and ANSI N14.6 [12]. The transfer cask can be provided in either a Standard or Advanced configuration. Canister handling, fuel loading and canister closing are operationally identical for either transfer cask configuration.

The transfer cask provides biological shielding when it contains a loaded canister and is used for the vertical transfer of the canister between work stations and the concrete cask, or transport cask. Five transfer casks of either configuration, having different lengths, are designed to handle the five canisters of different lengths containing one of three classes of PWR fuel assemblies or two classes of BWR fuel assemblies. In addition, a Transfer Cask Extension may be used to extend the operational height, when using the standard transfer cask. This height extension allows a transfer cask designed for a specific canister class to be used with the next longer canister.

The transfer cask design incorporates a top retaining ring, which is bolted in place to prevent a loaded canister from being inadvertently lifted through the top of the transfer cask. The transfer cask has retractable bottom shield doors. During loading operations, the doors are closed and secured by door lock bolts/lock pins, so they cannot inadvertently open. During unloading, the doors are retracted using hydraulic cylinders to allow the canister to be lowered into a concrete cask for storage or into a transport cask. A typical transfer cask is shown in Figure 1.2-2. The principal design parameters of the transfer casks are shown in Table 1.2-7.

To minimize the potential for contamination of a canister or the inside of the transfer cask during loading operations in the spent fuel pool, clean water is circulated in the annular gap between the transfer cask interior surface and the canister exterior surface. Clean water is processed or filtered pool water, or any water external to the spent fuel pool that is compatible. The transfer cask has eight supply and two discharge lines passing through its wall. Normally, two of the lines are connected to allow clean water under pressure to flow into and through the annular gap to minimize potential for the intrusion of pool water when the canister is being loaded. Lines not used for clean water supply may be capped. The eight supply lines can also be used for the introduction of forced air at the bottom of the transfer cask to achieve cooling of the canister contents. This allows the canister to remain in the transfer cask for an extended period, if necessary, during canister closing operations.

Standard and Advanced Transfer Casks

The Standard and Advanced transfer casks are designed for lifting and handling in the vertical orientation only. The Standard transfer cask may be used to lift canisters weighing up to 88,000 pounds. The Advanced transfer cask is similar to the Standard transfer cask, except that the

Advanced transfer cask incorporates a trunnion support plate that allows the Advanced transfer cask to lift canisters weighing up to 98,000 pounds. The Standard and Advanced transfer casks have four lifting trunnions, which allow for redundant load path lifting. Both transfer casks incorporate a multiwall (steel/lead/NS-4-FR/steel) design, and both designs have a maximum empty weight of approximately 121,500 pounds. The Standard and Advanced transfer cask designs are shown in Drawing 790-560.

1.2.1.5 Auxiliary Equipment

This section presents a brief description of the principal auxiliary equipment needed to operate the Universal Storage System in accordance with its design.

1.2.1.5.1 Transfer Adapter

The transfer adapter is a carbon steel table that is positioned on the top of the Vertical Concrete Cask or the Universal Transport Cask and mates the transfer cask to either of those casks. It has a large center hole that allows the Transportable Storage Canister to be raised or lowered through the plate into or out of the transfer cask. Rails are incorporated in the transfer adapter to guide and support the bottom shield doors of the transfer cask when they are in the open position. The transfer adapter also supports the hydraulic system and the actuators that open and close the transfer cask bottom doors.

1.2.1.5.2 Air Pad Rig Set

The air pad rig set (air pad set) is a commercially available device, sometimes referred to as an air pallet. When inflated, the air pad rig set lifts the concrete cask by using high volume air flow. The air pads employ a continuous, regulated air flow and a control system that equalizes lifting heights of the four air pads by regulating compressed air flow to each of the air pads. The compressed air supply creates an air film between the inflated air cushion and the supporting surface. The thin film of air allows the concrete cask to be lifted and moved. Once lifted, the cask can be moved by a suitable towing vehicle, such as a commercial tug or forklift.

1.2.1.5.3 Automatic Welding System

The automatic welding system consists of commercially available components with a customized weld head. The components include a welding machine, a remote pendant, a carriage, a drive motor and welding wire motor, and the weld head. The system is designed to make at least one weld pass automatically around the canister after its weld tip is manually positioned at the proper location. As a result, radiation exposure during canister closure is much less than would be incurred from manual welding.

1.2.1.5.4 Draining and Drying System

The draining and drying system consists of a suction pump and a vacuum pump. The suction pump is used to remove free water from the canister cavity. The vacuum pump is a two-stage unit for drying the interior of the canister. The first stage is a large capacity or "roughing" pump intended to remove free water not removed by the suction pump. The second stage is a vacuum pump used to evacuate the canister interior of the small amounts of remaining moisture and establish the vacuum condition.

1.2.1.5.5 Lifting Jacks

Hydraulic jacks are installed at jacking pads in the air inlets at the bottom of the concrete cask to lift the cask so that the air pad set can be installed or removed. Four hydraulic jacks are provided, along with a control panel, an electric hydraulic oil pump, an oil reservoir tank and all hydraulic lines and fittings. The jacks are used to lift the cask approximately three inches. This permits installation of the air pad rig set under the concrete cask.

1.2.1.5.6 Heavy-Haul Trailer

The heavy-haul trailer is used to move the Vertical Concrete Cask. A special trailer is designed for transport of the empty or loaded concrete cask. The design incorporates a jacking system that facilitates raising the concrete cask to allow installation of the air pad set used to move the cask onto the storage pad. The trailer incorporates both reinforcing to increase the trailer load-bearing area and design features that reduce its turning radius. However, any commercial double-drop-frame trailer having a deck height approximately matching that of the storage pad could be used.

1.2.1.5.7 Transporter

A cask transporter may also be used to move an empty or loaded Vertical Concrete Cask. The typical design incorporates a vertical lifting system that raises the concrete cask using the Vertical Concrete Cask lifting lugs. The transporter may be a self-propelled, towed or pushed design.

1.2.1.5.8 Helium Leak Test Equipment

A helium leak detector and leak test fixtures are required to verify the integrity of the welds of the canister shield lid. The helium leak detector is the mass spectrometer type.

9.2 Maintenance Program

This section presents the maintenance requirements for the UMS® Universal Storage System and for the transfer cask.

9.2.1 UMS® Storage System Maintenance

The UMS® Universal Storage 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 UMS® Universal Storage System has no valves, gaskets, rupture discs, seals, or accessible penetrations. Consequently, there is no maintenance associated with these types of features.

The routine thermal performance surveillance requirements for a loaded UMS® System are described in the Technical Specifications of Appendix A, Limiting Condition for Operation (LCO) 3.1.6.

Per the LCO, an initial verification of the concrete cask's thermal performance is completed by taking temperature measurements, per Surveillance Requirement (SR) 3.1.6.2, between 5 and 30 days following the start of storage operations.

Following the initial temperature measurements, 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 or placed in a safe condition as specified in the LCO.

In the event of any off-normal, accident or natural phenomena event, which could lead to the blockage of the concrete cask’s inlets and outlets, full vent blockage shall be removed within 24 hours, and any partial blockage shall be corrected to restore the cask to operable status in accordance with LCO 3.1.6.

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 UMS® systems in service shall be implemented. The Vertical 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 grouting of the area in accordance with the grout manufacturer's recommendations.
- Visually inspect accessible exterior coated carbon steel surfaces including lifting lug assemblies, if installed, for loss of coating, corrosion or other damage. The maintenance and 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. The licensee shall identify, evaluate and select acceptable coatings for use in routine maintenance of concrete cask external carbon steel surfaces.
- Visually inspect lid bolts for presence of corrosion. Excessively corroded or missing bolting shall be replaced with approved spare parts.
- Visually inspect the attachment hardware and the integrity of the inlet and outlet screens. Damaged or missing components shall be repaired or replaced with approved spare parts.
- 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 UMS® 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-UMS 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 9.2.1.

9.2.2 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 [8]. 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% (or 150% for facilities not implementing single-failure-proof lifting) 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 and critical areas.

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-UMS 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 9.2.2.

9.2.3 Required Surveillance of First Storage System Placed in Service

For the first Universal Storage System placed in service with a heat load equal to or greater than 10 kW, the canister is loaded with spent fuel assemblies and the decay heat load calculated for that canister. The canister is then loaded into the vertical concrete cask, and the cask's thermal performance is evaluated by measuring the ambient and air outlet temperatures for normal air flow. The purpose of the surveillance is to measure the heat removal performance of the Universal Storage System and to establish baseline data. In accordance with 10 CFR 72.4, a letter report summarizing the results of the surveillance and evaluation will be submitted to the NRC within 30 days of placing the loaded cask on the ISFSI pad. The report will include a comparison of the calculated temperatures of the NAC-UMS® system heat load to the measured temperatures. A report is not required to be submitted for the NAC-UMS® systems that are subsequently loaded, provided that the performance of the first system placed in service with a heat load ≥ 10 kW, is demonstrated by the comparison of the calculated and measured temperatures.

NAC's "Report on the Thermal Performance of the NAC-UMS® System at the Palo Verde Nuclear Generating Station (PVNGS) Independent Spent Fuel Storage Installation" [10] dated May 30, 2003, was transmitted to the NRC by Arizona Public Service on June 4, 2003, in accordance with the requirements of NAC-UMS® Technical Specification A 5.3, "Special Requirements for the First System Placed in Service," and in compliance with 10 CFR 72.4. The report concludes that the measured temperature data demonstrates that the thermal models and analysis results reported in the NAC-UMS® FSAR correctly represent the heat transfer characteristics of the storage system.

C3.0 NEW UFSAR SECTION 14

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-UMS 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-UMS 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 additional Aging Management Activities (AMAs) to address potential degradation during the period of extended operation. Tables 14.3-1 through Table 14.3-4 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.

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14.2 Time-Limited Aging Analysis

A comprehensive review was conducted of the NAC-UMS 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-UMS 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-UMS design.

As part of the CoC application for renewal, TLAAs have been prepared and incorporated into the NAC-UMS design bases for those in-scope SSCs. The additional TLAAs include: (1) Fatigue Evaluation of NAC-UMS System Components for Extended Storage; (2) Corrosion Analysis of NAC-UMS Steel Components for Extended Storage; (3) Aging Analysis for NAC-UMS System Neutron Absorber and Neutron Shield Components (Storage/Transfer), and (4) Thermal Aging Evaluation for Type 17-4PH Stainless Steel Support Disks for NAC-UMS PWR Fuel Basket. 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-UMS CoC Renewal Application [Ref. 14.6.8].

14.2.1 Fatigue Evaluation of NAC-UMS Storage System Components for Extended Storage [Ref. 14.6.2]

The potential fatigue of the NAC-UMS SSCs (e.g., canisters and fuel baskets) were evaluated in a TLAA for service conditions over the period of extended operation. The NAC-UMS 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-UMS canisters and fuel baskets do not require fatigue analysis for cyclic service for 60-years of extended storage conditions, TLAA has been prepared documenting why that analyses are not required.

14.2.2 Corrosion Analysis of NAC-UMS VCC Steel Components for Extended Storage [Ref. 14.6.3]

The TLAA evaluated the general corrosion of NAC-UMS 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.

The structural evaluation of the VCC for dead load, live load, flood, tornado wind, and seismic loading did not take any structural credit for the VCC steel inner shell, and therefore, it is concluded that any reduction in the VCC inner shell 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 VCC concrete shell due to tornado missile impacts did not take any structural credit for the VCC steel inner shell, 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 prevent missile perforation.

The structural evaluation of the VCC assembly for the VCC 24-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 inner shell, 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 1.67-inch, resulting in a 28% reduction of the air inlet cross-section area, which is bounded by the half inlets blocked condition. Furthermore, since the deformation of the corroded pedestal ring is much less than the 6-inch height of the air inlet opening, the weldment plate (and canister) will not "bottom-out". Due to reduced stiffness of the pedestal, the canister acceleration loads will be lower than those for calculated based on the nominal plate thicknesses.

The structural evaluation of the 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 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 NAC-UMS Neutron Absorber and Neutron Shield Components (Storage/Transfer) [Ref. 14.6.4]

NAC-UMS 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 $1x10^{-9}$ 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-UMS 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.

14.2.4 Thermal Aging Evaluation for Type 17-4PH Stainless Steel Support Disks for NAC-UMS PWR Fuel Basket [Ref. 14.6.2, Appendix A]

Thermal aging of the 17-4 PH precipitation-hardened stainless steel fuel basket support disks is dispositioned this Section. Based on the discussions in the study by Olender et. al (Ref. 14.6.18), applicable conditions for the 17-4PH support disks of the NAC-UMS PWR basket were evaluated, which include initial heat treatment condition, service temperature, operating environment, and stress level. As shown in the following discussion, thermal aging will not compromise the safety function of the 17-4PH support disks.

Material Condition

The support disks are made of 17-4PH stainless steel per ASME SA-693, Type 630, heat treated to condition H1150. In the recommended actions in the study by Olender et. al, the H1150 is considered optimally heat treated, if the design allows it, which is the case for this application.

Service Temperature

The study by Olender et. al provides guidance for assessing the potential for embrittlement for the 17-4PH stainless steel for operating temperatures between 470°F and 600°F. The maximum temperature of 601°F of the 17-4PH support disks corresponds to the design basis heat load case for the normal condition of storage. This temperature occurs at a very limited region at the center of the middle disk of the basket. The average temperature of all the support disks is 397°F, with temperature at the center of the disks ranging from 294°F to 601°F and the temperature at the disk edges ranging from 165°F to 400°F. Note that this temperature profile corresponds to the design basis heat load condition. Also note that the operating temperatures for all the system components continually decrease during the lifetime of the storage as a result of fuel decay.

Operating Environment

The cases of the failures of 17-4PH components observed and described in the study by Olender et. al occurred in active components such as valves or subcomponents of valves, which were in a pressure boundary and were subjected to dynamic loading. This indicates that the failures are associated with large primary stresses. Additionally, much of the operating experiences described in the study indicate an element of corrosion, SCC or overloading contributing to the observed failure.

During the long-term storage condition, the basket support disks maintain the locations of each fuel tube for criticality control. The support disks are individually supported at eight tie-rod locations and are subjected to static load from the self-weight of the disks only. The support disks are not subjected to dynamic or cyclic loads.

Since the canisters are backfilled with helium, the support disks are in an inert environment, not an aggressive chemical environment as discussed in the study by Olender at. el. Also, much of this operating experience in the study is in a system pressure boundary or subject to dynamic loads on an active component. The loading condition and the operating environment for the support disks in the long-term storage conditions are static and mild. This type of operating environment would not challenge the safety function of the support disks exposed to any potential thermal aging effects.

Stress Level

During the long-term storage condition, the support disks are in a horizontal position and individually supported at eight tie-rod locations. Each support disk is subjected to a static inertial load corresponding to its self-weight only, resulting in minimal stresses in the disk. As shown in Table 3.4.4.1-12 of the renewal application, the maximum stress intensity for the primary membrane plus primary bending ($P_m + P_b$) stress is 0.8 ksi. Using the allowable stress as defined by the ASME Code Subsection NG is 52.7 ksi, which is based on a conservative temperature of 800°F, the minimum margin of safety for the support disks is 64.8. Since the support disks are under such a low stress level, any thermal aging effect on this favorably heat-treated material would have a negligible effect on the structural function of the support disk.

Conclusion of Evaluation

The 17-4PH support disks for NAC-UMS PWR system are in a service where some portion of them can be exposed to operating temperatures between 470°F and 600°F, which can potentially cause embrittlement. However, the component failures described in the study by Olender at. el. are associated with the combined conditions of high stresses due to dynamic or cyclic loads and elevated operating temperatures. Therefore, based on the discussion above, any thermal aging effect will not adversely impact the safety function of the 17-4PH support disks because of the following characteristics and conditions.

- The support disks have been heat treated to an optimum condition to minimize the susceptibility of the material to thermal embrittlement.
- Only a limited portion of the disks are subjected to temperatures above 470°F and the average disk temperature is below 400°F for normal conditions of storage (Note that all disks are subjected to negligible static primary stresses).
- The disks are stainless steel materials in an inert operating environment. The loading condition and the operating environment for the support disks in the long-term storage conditions are static and mild.
- The support disks are subjected to insignificant static loads (self-weight only) with a very low stress level.

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 14.3-4. 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-UMS System in-scope SSCs include the following:

1. Aging Management Program for Localized Corrosion and Stress Corrosion Cracking (SCC) of Welded Stainless-Steel Transportable Storage Canisters (TSCs)
2. Aging Management Program for Internal Vertical Concrete Casks (VCC) - Metallic Components Monitoring
3. Aging Management Program for External Vertical Concrete Casks (VCC) - Metallic Components Monitoring
4. Aging Management Program for Reinforced Vertical Concrete Casks (VCC) - Structures Concrete Monitoring
5. Aging Management Program for Transfer Casks (TFRs) and Transfer Adapters
6. Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment

The AMPs for the NAC-UMS Systems are provided in Tables 14.3-5 through 14.3-10.

Table 14.3-1 Aging Management Activity Results - NAC-UMS 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	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
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
	Stainless Steel (Welded)	SH	Stress Corrosion Cracking	Cracking	TSC Localized Corrosion and SCC AMP
Port Cover	Stainless Steel	FE	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
Fuel Basket Bottom Weldments	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code

Table 14.3-1 Aging Management Activity Results - NAC-UMS Transportable Storage Canister (TSC) and Fuel Basket (FB) (continued)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Fuel Basket Top Weldments	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
Fuel Basket Support Disks	Steel	HE	Fatigue	Cracking	TLAA per Design Code
	Stainless Steel (17-4 PH)	HE	Fatigue	Cracking	TLAA per Design Code
		HE	Thermal Aging	Loss of Fracture Toughness / Loss of Ductility	TLAA
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
DFC Lid Plate, Lid Bottom Plate and Dowel Pins	Stainless Steel	HE	Fatigue	Cracking	TLAA per Design Code
DFC Bottom and 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 and Support Ring	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-UMS 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
			Pitting and Crevice Corrosion	Loss of Material	TLAA and External VCC Metallic Monitoring AMP
Top Flange	Steel	E-C	General Corrosion	Loss of Material	TLAA
			Pitting and Crevice Corrosion	Loss of Material	TLAA
Base Weldment (Baffle Weldment) 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	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

Table 14.3-2 Aging Management Activity Results - NAC-UMS Vertical Concrete Cask (VCC) (continued)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Weldment Base Plate	Steel	SH	Pitting and Crevice Corrosion	Loss of Material	TLAA
			General Corrosion	Loss of Material	TLAA
Base Weldment Bottom 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
		EC	General Corrosion	Loss of Material	TLAA
			Pitting and Crevice Corrosion	Loss of Material	TLAA
Nelson Stud	Steel	E-C	General Corrosion	Loss of Material	TLAA
			Pitting and Crevice Corrosion	Loss of Material	TLAA
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

Table 14.3-2 Aging Management Activity Results - NAC-UMS Vertical Concrete Cask (VCC) (continued)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
VCC Lid	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	TLAA and External VCC Metallic Monitoring AMP
		SH	General Corrosion	Loss of Material	TLAA
			Pitting and Crevice Corrosion	Loss of Material	TLAA
Shield Plug Assembly	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
	NSP/NSC (NS-4-FR/ NS-3)	FE	Thermal Aging	Loss of Fracture Toughness/Loss of Ductility	TLAA
			Radiation Embrittlement (NS-4-FR only)	Cracking	TLAA
			Boron Depletion (NS-4-FR only)	Loss of Shielding Effectiveness	TLAA
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 14.3-2 Aging Management Activity Results - NAC-UMS Vertical Concrete Cask (VCC) (continued)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Concrete Shell	Concrete	OD	Reaction with Aggregates	Cracking	Reinforced VCC Structures AMP
				Loss of Strength	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
			Salt Scaling	Loss of Material (Spalling, Scaling)	Reinforced VCC Structures AMP
Inlet Vent Supplemental Shield Assemblies or Shield Bars	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

Table 14.3-2 Aging Management Activity Results - NAC-UMS Vertical Concrete Cask (VCC) (continued)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Lift Anchor Lug, Base Plate and Spacer Plate	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
		E-C	General Corrosion	Loss of Material	TLAA
			Pitting and Crevice Corrosion	Loss of Material	TLAA
Lifting Anchor Hardware – Nut and Washer	Steel	OD	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
Lifting Anchor Rebar and Threaded Rebar	Steel	E-C	General Corrosion	Loss of Material	TLAA
			Pitting and Crevice Corrosion	Loss of Material	TLAA
VCC Lid Bolts	Stainless Steel	OD	Galvanic Corrosion	Loss of Material	TLAA and External 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-UMS Transfer Cask (TFR) and Transfers Adapter

Subcomponent	Material[1)]	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 (Transfer Cask Body and Shield Doors)	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

Table 14.3-3 Aging Management Review Results - NAC-UMS Transfer Cask (TFR) and Transfer Adapters (continued)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Shield Door Rail	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
			Wear	Loss of Material	Transfer Cask AMP
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 Bolt	Stainless Steel (Ferritic)	OD	Galvanic Corrosion	Loss of Material	Transfer Cask AMP
Transfer Cask Extension	Steel	OD	General Corrosion	Loss of Material	Transfer Cask AMP
			Pitting and Crevice Corrosion	Loss of Material	Transfer Cask AMP
Shield Door Assembly	Steel	OD	Pitting and Crevice Corrosion	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
			General Corrosion	Loss of Material	Transfer Cask AMP
Wear Strip	Stainless Steel (Nitronic 30)	OD	Wear	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 Aging Management Review Results – NAC-UMS Spent Fuel Assemblies (SFAs)

Subcomponent	Material (1)	Storage Operation Environment (2)	Aging Mechanism	Aging Effect	Aging Management Activities Required
Fuel Rod Cladding	Zirconium-based Alloy	HE	Hydride Reorientation (HBU SNF only)	Cladding Breach/Structural Failure	High-Burnup Fuel Monitoring and Assessment AMP
			Thermal Creep (HBU SNF only)	Changes in Dimensions	High-Burnup Fuel Monitoring and Assessment 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 14.3-5
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.

Table 14.3-5
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)	Frequency • Baseline inspection at beginning of the period of extended operation • Subsequently every 10 years for TSCs without detection of indications of major corrosion degradation or SCC • Subsequently every 5 years for TSCs with detection of major indications of corrosion degradation or detection(s) of SCC 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 the 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

Table 14.3-5
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	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) 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 (2-inch) 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

Table 14.3-5
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
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
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-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
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 TSC Systems</u> 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 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 14.3-6
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. • 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.

Table 14.3-6
AMP-2 - Aging Management Program for Internal Vertical Concrete Casks (VCC) Metallic Components Monitoring (continued)

AMP Element	AMP Description
4. Detection of Aging Effects (continued)	<u>Timing</u> These are opportunist inspections conducted in conjunction with TSC inspections. This inspection is performed when the TSC inspection is conducted.
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 the 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.

Table 14.3-6
AMP-2 - Aging Management Program for Internal 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

Table 14.3-6
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 the subject inspection report in Appendix E. 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 14.3-7
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, 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. <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.

Table 14.3-7
AMP-3 - Aging Management Program for External Vertical Concrete Casks (VCC) Metallic Components Monitoring (continued)

AMP Element	AMP Description
4. Detection of Aging Effects	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.

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 or lift lug hardware and loose bolts on VCC lid and lifting lug bolting, 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 14.3-7

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

AMP Description	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 14.3-8
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 structures 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. • Increase in Gamma dose rates exceeding LCO A 3.2.2 levels.
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 14.3-8
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 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 new tier 2 findings may be accepted after review by designated responsible-in-charge engineer • All new tier 3 findings must be reviewed by the designated responsible-in-charge engineer and are subject to further evaluations as appropriate for the finding. New tier 3 indications or any other indication which could potentially increase dose rates shall be inspected with Gamma Dose rate measurements and verified to be less than LCO A 3.2.2 acceptance criteria. 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) • Increases in Gamma Dose rates exceeding LCO A 3.2.2 acceptance criteria
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 14.3-8
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. Inspection OE for NAC-UMS and NAC-MPC VCC Concrete Structures 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 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 14.3-9
AMP-5 - Aging Management Program for Transfer Casks (TFR) and Transfer Adapters

<table>
<tr><th>AMP Element</th><th>AMP Description</th></tr>
<tr><td>1. Scope of Program</td><td>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.</td></tr>
<tr><td>2. Preventive Actions</td><td>This program is for condition monitoring and does not include preventative actions.</td></tr>
<tr><td>3. Parameters Monitored/ Inspected</td><td>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.</td></tr>
<tr><td>4. Detection of Aging Effects</td><td><u>Method or Technique</u>

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 or their equivalent.
Personnel performing visual examinations per this AMP shall meet the qualification requirements of IWE-2330(b) or their equivalent.</td></tr>
</table>

Table 14.3-9
AMP-5 - Aging Management Program for Transfer Casks (TFR) and Transfer Adapters (continued)

Element	Description
4. Detection of Aging Effects (continued)	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. Data Collection Documentation of the inspections required by this AMP, shall be added to 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.

Table 14.3-9
AMP-5 - Aging Management Program for Transfer Casks (TFR) and Transfer Adapters (continued)

Element	Description
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.
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.

Table 14.3-10
AMP-6 - Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment

AMP Element	AMP Description
1. Scope of Program	The High-Burnup (HBU) Fuel Aging Management Program (AMP) is a generic program applicable to all licensees storing uncanned HBU (≥ 45 GWd/MTU) Fuel Assemblies (FAs) in the NAC-UMS System [(1)] beginning in March 2009. HBU FAs stored in the NAC-UMS System are limited by Technical Specification to an assembly average burnup of ≤ 60 GWd/MTU. The maximum HBU FA burnup loaded into a NAC-UMS System had a nominal burnup of < 55 MWd/MTU and was loaded on July 18, 2014. The HBU FAs have Zirconium alloy fuel cladding (e.g., Zirc-4, ZIRLO, low-tin Zirc-4, and M5). All HBU FAs will be stored in a high-purity Helium atmosphere and are bounded by the by the temperature limits of ISG-11, Revision 3. The program is to manage the factors that could affect the ability to comply with 10 CFR 72.122(h)(1) including fuel cladding temperature, fuel cladding breach, assembly distortion, residual moisture after drying, changes in hydride structure of the cladding, and cladding creep.
2. Preventive Actions	This program is for condition monitoring and does not include preventative actions. During the initial loading operations of the NAC-UMS Transportable Storage Canisters (TSCs) the design bases and Certificate of Compliance (CoC) and Technical Specifications (TS) require that the fuel be stored in a dry inert environment. TS Limiting Condition of Operation (LCO) A3.1.2 “Canister Vacuum Drying Pressure” demonstrates that the TSC cavity is dry by maintaining a cavity absolute pressure less than or equal to 10 torr for 10 minutes with the TSC isolated from the vacuum pump. Following the dryness verification, the TSCs are then re-evacuated to ≤ 3 torr, backfilled with 99.9% pure helium to atmospheric, then re-evacuated to ≤ 3 torr for final helium backfill in accordance with TS LCO A3.1.3 “Canister Helium Backfill Pressure”. These two TS requirements ensure that the HBU fuel is stored in a dry and inert environment, thus preventing cladding degradation due to oxidation mechanisms. TS LCO A3.1.1 “Canister Maximum Time in Vacuum Drying” prescribes times that the helium environment be established after commencing TSC draining for design bases heat loads. HBU fuel contents loaded with total heat load of less than 17.4 kW had 33 hours to establish a final helium backfill. The NAC-UMS TSCs are loaded and processed in compliance within the temperature limits of ISG-11, Revision 3.

[1] HBU fuel assemblies (maximum burnup < 50 GWd/MTU) were loaded at Maine Yankee Atomic Power Plant (MY) in specially designed NAC-UMS damaged fuel cans (DFCs). A maximum of four MY HBU fuel assemblies in DFCs were preferentially loaded in four corner locations of the 24 PWR fuel basket in accordance with CoC Technical Specifications, Appendix B requirements. Therefore, this AMP is not applicable to MY canned HBU fuel in NAC-UMS Systems.

Table 14.3-10
AMP-6 - Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment (continued)

AMP Element	AMP Description
3. Parameters Monitored/ Inspected	The surrogate demonstration program, e.g., DOE/EPRI's HDRP, parameters monitored and/or inspected will include: • Fuel cladding temperature • Cavity gas temperature, pressure, and composition The monitoring of the above parameters supplemented by physical examination of should be able to provide confirmation if: • The models of degradation phenomena used for 20-year predictions can be used for the TLAA beyond 20 years. • The condition of the fuel, after an appropriately long period of storage, does not degrade. • New degradation mechanisms are not being exhibited.
4. Detection of Aging Effects	Since limited AMP action can be taken inside a sealed TSC, this program relies on the surrogate monitoring inspections of the DOE's HBU Dry Storage Cask Research and Development Project (HDRP) to verify no unexpected aging effects or to identify aging effects if they occur.
5. Monitoring and Trending	Monitoring and trending methods will be used to: • Assess information/data from the HDRP or from other sources (such as testing or research results and scientific analyses) when it becomes available. • The licensees will monitor, evaluate, and trend the information via their operating experience program and/or CAP to determine what actions should be taken to manage fuel and cladding performance, if any. • Formal evaluations (Tollgates) of the aggregate information from the HDRP and other available domestic or international operating experience (including data from monitoring and inspection programs, NRC-generated communications, and other information) will be performed at specific points in time during the period of extended operation, as delineated in Table 14.5-2 of the NAC-UMS FSAR. • If any of the acceptance criteria of Element 6 are not met, the licensee must conduct additional assessments and implement appropriate corrective actions per Element 7.

Table 14.3-10
AMP-6 - Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment (continued)

AMP Element	AMP Description
6. Acceptance Criteria	If any of the following fuel performance criteria are not met in the HDRP, a corrective action is required: • Cladding Temperature – The maximum cladding temperature measured is less than or equal to that predicted by the thermal analysis methods in the UFSAR for the affected TSCs • Cavity Gas Temperature – The average cavity gas temperature measured is less than equal to the that predicted by the thermal analysis methods in the FSAR for the affected TSCs • Cavity Gas Pressure – The cavity gas pressure measured should correspond to the gas pressure predicted for the thermal conditions and corresponding to < 1.0% failed fuel rods. • Cladding Creep – Total creep strain extrapolated to the to the total storage duration based on the best fit to the data accounting for initial condition uncertainty shall be less than 2.5% • Hydrogen content – Maximum hydrogen content of the cover gas over the approved storage period should be extrapolated from the gas measurements to be less than the design-bases limit for hydrogen content for the NAC-UMS Storage System of < 4% of free volume. • Drying –The moisture content in the TSC, accounting for measurement uncertainty, should be less than the expected upper-bound moisture content of < 0.5 gm-mole. • Fuel condition/performance–nondestructive examination (e.g., fission gas analysis) and destructive examination (e.g., to obtain data on creep, fission gas release, hydride reorientation, cladding oxidation, and cladding mechanical properties) confirms the design-bases fuel condition of ≤ 1% rod failure per the analyzed fuel configuration considered in the NAC-UMS System FSAR and approved design bases. • Fuel Rod Breach – Fission gas analysis shall not indicate more than 1.0% of fuel rod cladding breaches.

Table 14.3-10
AMP-6 - Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment (continued)

AMP Element	AMP Description
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). Corrective actions should be implemented if data from the HDRP or other sources of information indicate that any of the HDRP acceptance criteria are not met. If any of the acceptance criteria are not met, the licensee will: • Assess fuel performance (impacts on fuel and changes to fuel configuration), including any consequences of above-design-basis moisture levels on potential degradation of the fuel assembly. • Assess the design-bases safety analyses, considering degraded fuel performance (and any changes to fuel configuration), to determine the ability of the NAC-UMS System to continue to perform its intended functions under normal, off-normal, and accident conditions. The licensee will determine what corrective actions should be taken to: • Manage fuel performance, if any • Manage impacts related to degraded fuel performance to ensure that all intended functions for the NAC-UMS are met In addition, the licensee will obtain the necessary NRC approval in the appropriate licensing/certification process for modification of the design bases to address any conditions outside of the approved design bases.
8. Confirmation Process	The confirmation processes will be commensurate with the general 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 describes or references procedures to: • Determine follow-up actions to verify effective implementation of corrective actions • Monitor for adverse trends due to recurring or repetitive finding or observations.

Table 14.3-10
AMP-6 - Aging Management Program for NAC-UMS High-Burnup Fuel Monitoring and Assessment (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. The HBU Fuel Tollgate program described in FSAR Chapter 14, Table 14.5-2 will reference and evaluate applicable operating experience on a best-effort basis, including: • Internal and industrywide condition reports. • Internal and industrywide corrective action reports. • Vendor-issued safety bulletins. • NRC Information Notices and NUREGs • IAEA Publications on OE for HBU Fuel Performance • Applicable DOE or industry initiatives (e.g., HDRP). • Applicable research (e.g., Oak Ridge National Laboratory studies on bending responses of the fuel, Argonne National Laboratory and Central Research Institute of Electric Power Industry studies on hydride reorientation effects). The review of OE clearly identifies any HBU fuel degradation as either age related, or event driven, with proper justification for that assessment. Past operating experience supports the adequacy of the HDRP. NAC-UMS general licensees have no specific OE on the performance of HBU fuel, in dry storage to date.

14.4 Retrievability

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 [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-UMS 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-UMS 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-UMS canister to the approved and NRC certified NAC-UMS transport cask system (CoC No. 71-9270)) [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]), and limited high burnup (≥ 45,GWd/MTU) (loaded in DFCs) zircaloy alloy clad PWR SNF assemblies are stored in the NAC-UMS 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-UMS canister to be lifted and transferred to a NAC-UMS transport cask. During the AMR, the appropriate NAC-UMS 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-UMS transport cask for off-site transport.

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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 UMS 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 holder.

14.5.1 Tollgate Assessments by General Licensees

As detailed in Tables 14.5-1 and 14.5-2, during the twenty-fifth calendar year following a UMS dry storage system or a UMS dry storage system with HBU fuel (uncanned) beginning STORAGE OPERATIONS, the general licensee shall conduct and document an initial tollgate assessment of 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 and/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 Tables 14.5-1 and 14.5-2 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 14.3-5 through 14.3-10 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; transfer casks and transfer adapters; and High-Burnup Fuel Monitoring and Assessment. 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 14.3-5 through 14.3-9 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-UMS 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

14.5.2.3 HBU Fuel Monitoring and Assessment Tollgates

Different guidance has been developed for the AMP for High-Burnup (HBU) Fuel Monitoring and Assessment (Table 14.3-10) and is described in Table 14.5-2. Table 14.5-2 defines specific tollgates for the HBU Fuel AMP. If at any tollgate the information available is insufficient to demonstrate the ability of the HBU Fuel Assemblies (FAs) to perform their intended functions through the period to the next tollgate, then the general licensee shall:

1. Shorten the interval to the next tollgate and
2. Develop a plan and timeline to obtain sufficient information to demonstrate the ability of the HBU FAs to perform their intended functions.

If at any tollgate the information assessed indicates that acceptance criteria may not be met, then the general licensee shall:

1. Assess fuel performance,
2. Assess design basis safety analysis to determine if the UMS storage system can continue to perform its intended safety function under normal, off-normal, and accident conditions,
3. Identify corrective actions, if any, to be taken to manage fuel performance and ensure all intended functions of the UMS storage system can be met,
4. Obtain NRC approval for any modifications to the approved licensing basis as a result of this assessment.

Table 14.5-1 Generalized Tollgate Process for AMPs

Tollgate	Home	Assessment
1	T_0+ 5 years [(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 years	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 years	Same as Tollgate 1 as informed by the results of Tollgates 1 and 2
4	T_0 + 30 years	Same as Tollgate 1 as informed by the results of Tollgates 1, 2, and 3

Note: [(1)] T_0 is 20 years after first UMS dry storage system began STORAGE OPERATIONS.

Table 14.5-2 Specific Tollgate Process for High Burnup Fuel

Tollgate	Home	Assessment
1	T_0 (1) + 5 years	Evaluate information from the following sources and perform a written assessment of the aggregate impact of the information, including but not limited to corrective actions required and the effectiveness of the HBU AMP. • Initial data on gas analysis and cladding temperature from the EPRI and DOE "High Burnup Dry Storage Cask Research and Development Project" (HDRP) [14.6.10] or an alternative program meeting the guidance in Interim Staff Guidance (ISG) 24 [14.6.16]. • Relevant, available published research into HBU fuel performance in dry storage and transport, including fuel swelling, ductile to brittle transition of cladding, and mechanical testing by DOE, the national laboratories, and others [for example, the Institute for Transuranium Elements (ITU, Karlsruhe), Japan Nuclear Energy Safety Organization (JNES), Commissariat a l'energie atomique et aux energies alternatives (CEA), Korea Atomic Energy Research Institute (KAERI)]. • Entries in the AMID for other dry storage systems. • NRC documents such as Information Notices, Generic Letters or License Event Reports LERs resulting from dry fuel storage or HBU fuel findings. • Relevant results of other domestic and international nuclear and nonnuclear research, inspection results and OE.
2	T_0 (1) + 10 years	Evaluate, if available, information obtained from the destructive and nondestructive examination of the fuel placed into storage in the HDRP along with other available sources of information. If the destructive examination data from the HDRP has not been obtained in time to support the assessment required by this tollgate, then the General Licensee will coordinate with the Certificate Holder to outline plans to obtain evidence to demonstrate that the fuel performance acceptance criteria in Table 14.3-10 continue to be met.
3	T_0 (1) + 20 years	Same as Tollgate 2, as informed by the results of Tollgates 1 and 2.
4	T_0 (1) + 30 years	Same as Tollgate 3, as informed by the results of Tollgates 1, 2 and 3.

Note: (1) T_0 is 20 years after the first UMS dry storage system with HBU fuel (uncanned and average-assembly burnup > 45 GWd/MTU) 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-UMS Storage System Components for Extended Storage, NAC Calculation No. 30013-2001, Revision 2

14.6.3 Time-Limited Aging Analysis (TLAA) for Potential Corrosion of the Steel Components in the NAC-UMS Storage System VCC Assembly for a Service Life of 60-Year, NAC Calculation No. 30013-2002, Revision 2

14.6.4 Aging Analysis for NAC-UMS Neutron Absorber and Neutron Shield Components (Storage -Transfer), NAC Calculation No. 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-UMS Transport Cask, Docket 71-9270, CoC No. 9270, Revision 5, October 24, 2017.

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-UMS CoC Renewal Application, dated August 2020.

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. "HBU Dry Storage Cask Research and Development Project Final Test Plan." DOE Contract No.: DE-NE-0000593. Palo Alto, California: Electric Power Research Institute. 2014.

14.6.11 EPRI Technical Report, TR-3002008193, Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless-Steel Canisters

14.6.12 ED20170040, NAC Memorandum, Technical Report: NAC-UMS and NAC-MPC ISFSI and Individual TSC Rankings Based on EPRI CISCC Criteria,

14.6.13 EPRI Technical Report, TR-3002005371, Susceptibility Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless-Steel Canisters for Dry Storage

14.6.14 ACI 349.3R-18, “Report on Evaluation and Repair of Existing Nuclear Safety-Related Concrete Structures.” American Concrete Institute. 2018.

14.6.15 ANSI N14.6, “Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500kg) or More for Nuclear Materials”

14.6.16 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”

14.6.17 ACI 349.3R-02, “Evaluation of Existing Nuclear Safety Related Concrete Structures (Reapp 2010)”

14.6.18 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.