Exelon Generation®

Mark D. Flaherty
Acting Site Vice President

Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657

410 495 5205 Office
410 495 5444 Fax
443 534 5476 Mobile
www.exeloncorp.com

mark.flaherty@exeloncorp.com

10 CFR 72.56

September 11, 2015

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Calvert Cliffs Nuclear Power Plant
Independent Spent Fuel Storage Installation, License No. SNM-2505
NRC Docket No. 72-8

Subject: Amendment Request No. 1 to Renewed Materials License No. SNM-2505 for Calvert Cliffs Specific Independent Spent Fuel Storage Installation - Second Request for Additional Information, Part 2

References:
1. Letter from G. H. Gellrich (Exelon) to Document Control Desk (NRC), dated March 26, 2013, License Amendment Request: High Burnup NUHOMS®-32PHB Dry Shielded Canister and Horizontal Storage Modules
2. Letter from J. M. Goshen (NMSS) to G. H. Gellrich (Exelon), dated June 23, 2015, Amendment Request No. 1 to Renewed Materials License No. SNM-2505 for Calvert Cliffs Specific Independent Spent Fuel Storage Installation - Second Request for Additional Information, Part 2 (TAC No. L24912)

Reference 1 submitted a license amendment request for the Calvert Cliffs Nuclear Power Plant site-specific independent spent fuel storage installation. The amendment, if approved, would authorize the storage of Westinghouse and AREVA Combustion Engineering 14x14 fuel in the NUHOMS-32PHB Dry Shielded Canister system. As part of their review, the NRC staff has requested additional information (Reference 2). Responses to the requested additional information are provided in Attachment (1).

Enclosures 3, 4, and 5 contain information that is proprietary to AREVA Inc., therefore, they are accompanied by an affidavit signed by AREVA, the owner of the information (Attachment 2). The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission, and addresses, with specificity, the consideration listed in 10 CFR 2.390(b)(4). Accordingly, it is requested that the information that is proprietary to

NMSS26

AREVA, Inc. be withheld from public disclosure. Non-proprietary versions of Enclosures 3, 4, and 5 are not available.

The additional information provided does not change the environmental assessment provided in Reference 1 and the categorical exclusion set forth in 10 CFR 51.22(c)(11) is still valid. There are no regulatory commitments contained in this correspondence.

Should you have questions regarding this matter, please contact Mr. Larry D. Smith at (410) 495-5219.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 11, 2015.

Respectfully,

Mark D. Flaherty
Acting Site Vice President

MDF/PSF/bjm

Attachment: (1) Second Request for Additional Information, Part 2
Enclosures: 1 USAR Draft Section 9.6
2 Drawings
3 PROPRIETARY Calculation NUH32PHB-0201
4 PROPRIETARY Calculation NUH32PHB-0202
5 PROPRIETARY Calculation NUH32PHB-0203
(2) AREVA Affidavit

cc: NRC Project Manager, Calvert Cliffs
NRC Regional Administrator, Region I
NRC Resident Inspector, Calvert Cliffs
S. Gray, MD-DNR
Director – NMSS
J. M. Goshen, NMSS

ATTACHMENT (1)

SECOND REQUEST FOR ADDITIONAL INFORMATION, PART 2

By letter dated March 26, 2014, Exelon Generation Company, LLC, submitted license amendment request (LAR) No. 1 to the U.S. Nuclear Regulatory Commission (NRC) for Renewed Materials License No. SNM-2505 for the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI). The NRC staff has requested the following information.

Chapter 5 Structural and Materials Evaluation

NRC RAI 5-1:

Provide all applicable drawings associated with the NUHOMS® 32PHB Dry Shielded Canister (DSC) system.

The staff received drawings in response to request for supplemental information (RSI) 5-7 (ADAMS Accession No. ML14202A337) but is unsure if we possess all of them in a readable form. For example, vendor DWG Nos. NUH32PHB-03-3 sheet 2 of 2 and NUH32PHB-30-4 sheet 2 of 2 are presented as DWG No. 84239SH0002. The staff does not possess sheet 1 of 2 for either of these vendor drawings. DWG No. NUH32PHB-30-4 sheet 2 of 2 contains details and sections, but without sheet 1 of 2, the staff is unaware of the component from which the details and sections are taken. Ensure DWG No. NUH32PHB-30-5 is among the included drawings. Additionally, calculation No. NUH32PHB-0201, NUHOMS® 32PHB, "Weight Calculation of DSC/TC System," contains sketches in the appendix that appear to be excerpts from drawings. One of these sketches is a list of American Society of Mechanical Engineers (ASME) code exceptions for the DSC and the basket, but additional notes and the title block with the drawing number are cut off. These code exceptions do not appear anywhere in the proposed revision to the updated safety analysis report (USAR). Although Regulatory Guide 3.62 states that drawings on 8 ½ x 11 inches are preferred, due to the amount of detail the staff prefers the drawings be submitted on 11 x 17 inches so that the details are legible.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to RAI 5-1:

The response was provided in Reference 1.

NRC RAI 5-2:

Provided justification why the new canister lead gamma shielding is now a Category C material not in accordance with NUREG/CR-6407 guidance.

Drawing parts list provided as part of response to request for additional information (RAI) (ADAMS Accession No. ML14288A127) states the new canister's lead gamma shielding is a category C material.

This information is needed to demonstrate compliance with 10 CFR 72.24 and 10 CFR 72.56.

CCNPP Response to RAI 5-2:

The response was provided in Reference 1.

NRC RAI 5-3:

Provide a scoping evaluation and aging management review (AMR) for the transfer cask with forced-air cooling modifications. Clarify any changes to the NRC-approved aging management program (AMP) for the transfer cask to ensure the AMP is adequate for managing age-related degradation of new subcomponents.

The licensee provided a summary of a scoping evaluation and aging management review for the HSM-HB and NUHOMS® 32PHB DSCs, but did not provide a similar discussion for the transfer cask (see pages 3 and 4 of the license amendment request (LAR)). The licensee stated that the transfer cask uses a new lid, which contains small openings around the periphery for air venting (see page 2 of the LAR). The licensee further states that the modified cask includes a 0.5 inch thick spacer disc with wedge shaped protrusions is installed at the bottom of the transfer cask to facilitate air flow coming through the ram access opening to the annular space around the DSC. The staff notes that drawing NUH-06-8021 also identifies other subcomponents important-to-safety, which are not discussed in the application. The staff requires a comprehensive scoping evaluation and aging management review (AMR) for all new subcomponents meeting criterions 1 and 2 per Section 2.4.2 of NUREG-1927. The staff evaluates the results of the scoping evaluation and aging management review to ensure that the NRC-approved AMP for the transfer cask is adequate for all new subcomponents of the transfer cask.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to RAI 5-3:

USAR Section 9.6.3.2 and Table 9.6-4 have been revised to address the Forced Cooling option for the Transfer Cask. No aging management program is required for the forced cooling components. This includes the forced cooling Top Cover Assembly which is Class B and the Wedges at the bottom of the Transfer Cask which are Class C. Both items are stainless steel and are stored in a sheltered environment. Both the lid and the wedges are classified as NITS passive components in accordance with Criterion 1 and 2 per Section 2.4.2 of NUREG-1927. Refer to Table 9.6-4 for classifications (Enclosure 1).

NRC RAI 5-4:

Provide all applicable drawings for the HSM-HB design and the transfer cask with forced-air cooling modifications. Ensure a bill of materials and safety classification are provided for all subcomponents.

The staff received drawings in the RSI 5-7 response (ADAMS Accession No. ML14202A387), but does not possess all of them in a readable form. For example, the HSM-HB door was modified and identified to be constructed of the same material as the HSM model. The staff reviewed drawing NUH-03-7102, which references two drawings for the HSM door that were not included in the response to RSI 5-7 (NUH-03-7106 "Door Type A" and NUH-037108 "Door Type B"). In another example, drawing NUH-06-8021 references drawing NUH-06-8003. The latter was not provided in the response to RSI 5-7 but it identifies the structural shell assembly incorporating the new spacer disc with wedge shaped protrusions for the forced-air cooling configuration. The staff requires all pertinent drawings (including bill of materials and safety classification) to ensure that the scoping evaluation and aging management review for the HSM-HB and the transfer cask is accurate, and the NRC-approved aging management programs (AMPs) are adequate for the new designs.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to RAI 5-4:

Enclosure 2 contains the requested drawings.

- NUH-03-7106 Door Type A- attached
- NUH-03-7108 Door Type B- attached
- NUH-06-8003- attached

NRC RAI 5-5:

Provide proposed changes to Chapter 9 of the ISFSI USAR in order to address aging management of the NUHOMS® 32PHB DSC, HSM-HB and modified transfer cask designs. The changes should include:

- *A summary of all supporting TLAAs provided in the license amendment request and responses to RAI-1 (ADAMS Accession No. ML14288A127).*
- *Revisions to indicate that approved aging management programs (AMPs) will be applicable to the new designs, or summaries of proposed changes to NRC-approved AMPs, and*
- *Revised Tables 9.6-1 through 9.6-4, "Aging Management Review Results". The listings in these tables should clearly differentiate subcomponents specific to the new designs in the license amendment.*

The supplemented renewal application (ADAMS Accession No. ML14267A065) included an "ISFSI Updated Safety Analysis Report Supplement and Changes" supplement. The renewed license requires the licensee to incorporate this supplement in Chapter 9 of the ISFSI USAR (see License Condition 18, ADAMS Accession No. ML14274A022). The licensee should provide proposed changes to the ISFSI USAR in order to properly document the aging management review, TLAAs and AMPs pertinent to the NUHOMS® 32PHB DSC, HSM-HB designs and transfer cask.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to RAI 5-5:

USAR Section 9.6 has been revised to include all new passive components for the modified transfer cask, the HSM-HB and the 32PHB DSC. It is contained in Enclosure 1.

The following report and calculations have been updated to include the HSM-HB, 32PHB and Transfer Cask with Forced Cooling Options:

- TN Calculation 10955-0201 Rev.1 – Enclosure 3
- TN Calculation 10955-0202 Rev. 1 – Enclosure 4
- TN Calculation 10955-0203 Rev.1 – Enclosure 5

TN Technical Report 10955-0101 Rev.1 is being finalized and will be provided by September 30, 2015.

NRC RAI 5-6:

Provide additional details on the criteria to include the HSM-HB and NUHOMS® 32PHB DSC components into the next lead canister inspection as noted in NUREG-1927, Appendix E (see page 4 of LAR).

The staff requests additional information on the mechanisms by which the licensee will make a determination for the need to conduct baseline inspection(s) on the HSM-HB and NUHOMS® 32PHB DSC. The licensee has stated that the new designs will be included in the scope of the approved Aging Management Programs, but did not indicate if a baseline inspection will be conducted on the new designs prior to their period of extended operation (including expected sample size). While this inspection would not occur until 20 years after the amendment is approved, the staff is interested in details on the mechanisms and provisions in the licensee's Corrective Action Program for ensuring that degradation in the new designs is properly baselined prior to the start of the period of extended operation. The staff notes that it has recently discussed draft staff guidance for the review of lead system inspection data, including criteria for lead system(s) selection (see ADAMS Accession No. ML15068A331).

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to RAI 5-6:

The basis for lead canister inspection is to inspect the canister for evidence of SCC. Industry reports have determined that deliquescence occurs on surfaces with temperatures starting at and below 80°C.

For a newly loaded 32PHB canister to cool sufficiently to allow inspection and to reach the optimal 80°C susceptible temperature for deliquescence requires a minimum of 20 years. When the 32PHB canister has reached this temperature, a surface inspection will be performed.

Currently the 32PHB is not licensed and is not stored in the HSM-HB modules. The 32PHB DSC is scheduled to be utilized for the August 2016 fuel storage campaign provided the 32PHB DSC is licensed to support the 2016 fuel storage moves.

TN Calculation 10955-0401 evaluated the required time for a fully loaded 32P to cool down to 80°C and 19kW as 35.5 years. The 32PHB has a higher heat load and would therefore require a longer storage time to reach the 80°C threshold. Therefore, the first possible inspection of a 32PHB inside the HSM-HB would conservatively be scheduled for 2026.

A reinspection of the baseline 24P canister is scheduled for 2019. Information from this inspection will be used for assessing the timely inspection of the 32PHB DSC.

Using the Corrective Action Program the inspection dates and parameters will be modified as necessary to support the future DSC inspections for the potential effects of CISCC. These parameters will include:

- Surface sampling of the 32PHB using salt smart (or equivalent) technology.
- Visual inspections of the HSM-HB and 32PHB utilizing remote visual cameras accessed through the upper vent.
- Remote temperature readings of the 32PHB through the front access door.

NRC RAI 5-7:

Provide details on how the data from the DOE-funded High Burnup Dry Storage Cask Research and Development Project" (HDRP) will be used to make a determination on safe storage of high burnup fuel up to 62 GWd/MTU.

The current maximum fuel assembly average burnup limit is 47,000 MWd/MTU) for the NUHOMS® 24P DSCs and 52,000 MWd/MTU for the NUHOMS® 32P DSCs. The proposed amendment would add a new maximum fuel assembly average burnup limit of 62,000 MWd/MTU for fuel stored in NUHOMS® 32PHB DSCs. The High Burnup Fuel Aging Management Program in Attachment 4 of the supplement to the license renewal application (ML14267A065) was approved per guidance in ISG-24. ISG-24 indicates that the maximum burnup of the fuel in the application should be less than the burnup of the fuel in the demonstration. If the burnup is higher than the fuel in the demonstration program, the applicant would provide evidence, based on characteristics of the fuel, that the demonstration fuel is reasonably characteristic of the stored fuel and the additional burn-up would not change the results determined by the demonstration. Per the latest DOE Cask Demo Design Loading Plan (ML15133A082), the maximum burnups are: 55.5 GWd/MTU (ZIRLO™ cladding), 53.5 GWd/MTU (M5® cladding), 50.6 GWd/MTU (Zircaloy-4 cladding), and 50 GWd/MTU (low-Sn Zircaloy cladding). The applicant is asked to justify that the demonstration results from lower burnup cladding will be characteristic for higher burnup cladding; more specifically, that the acceptance criteria in the High Burnup Fuel Aging Management Program is adequate for burnups up to 62 GWd/MTU.

This information is needed to demonstrate compliance with 72.122(h)(1) and 72.122(1).

CCNPP Response to RAI 5-7:

Figure 1 below provides the burnup distribution of fuel stored in the Calvert Cliffs spent fuel pool as of June 30, 2015. As can be seen, the majority of fuel with Zircaloy-4 or Zirlo cladding have assembly average burnups that are either bounded by, or only slightly exceed, that in the DOE Cask Demonstration Design Loading Plan. This occurs because the Westinghouse/Combustion Engineering (C-E) fuel rod design methods are only licensed for a peak fuel rod average burnup of 60 GWd/MTU for those cladding materials for C-E designed plants, as detailed in CENPD-404-P (see ML012670041), which effectively limits assembly average burnups to ~55-56 GWd/MTU. New batches of fuel with Zircaloy 4 or Zirlo cladding have not been loaded since 2010. AREVA fuel with M5 cladding has only been used at Calvert Cliffs since batches were loaded into the core in 2011. As a result, Figure 1 does not provide significant information on final discharge burnups for that cladding material. However, AREVA's fuel rod design method detailed in EMF-92-116PA (non-proprietary version can be found at ML003681063) is only licensed to a peak fuel rod average burnup of 62 GWd/MTU, which effectively limits assembly average burnup to < 58 GWd/MTU. Therefore, it is expected that the DOE Cask Demonstration results for M5 cladding will also be representative of the highest burnup AREVA fuel stored in the 32PHB DSCs at Calvert Cliffs. The few Lead Test Assemblies (LTAs) with very high burnup and/or alternative cladding materials or dimensions are not being targeted for storage in the 32PHB DSC at this time.

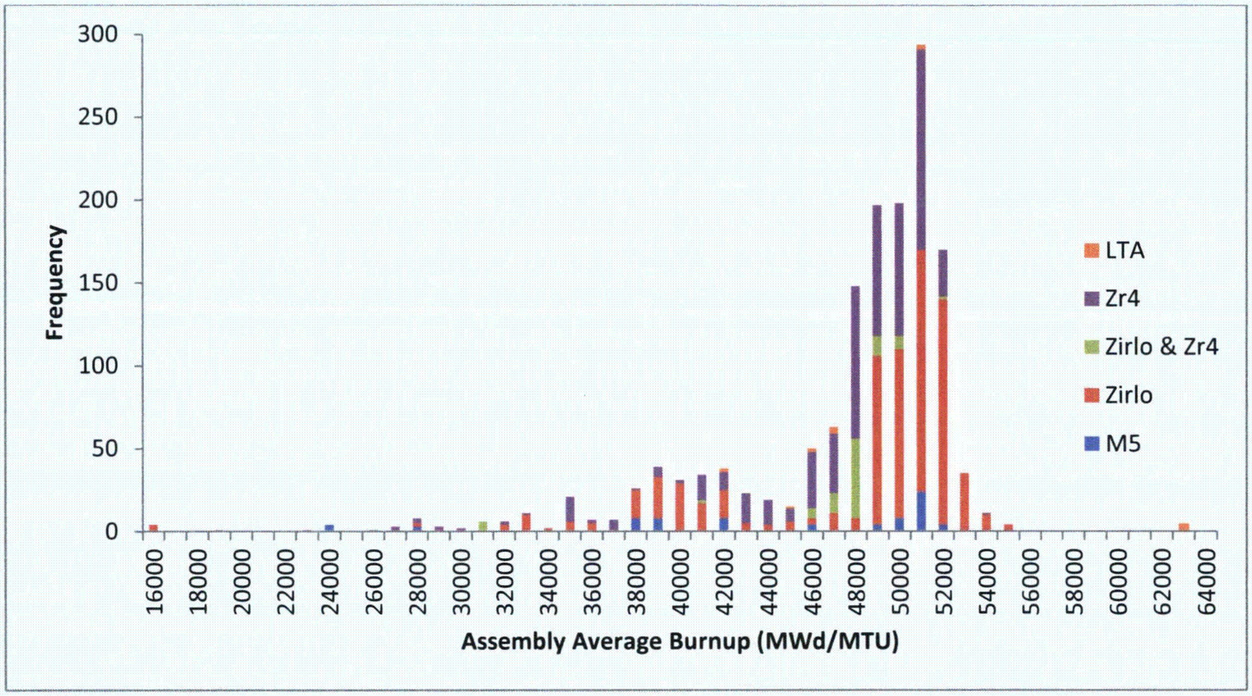

Figure 1 – Distribution of Burnup for Discharged Fuel at Calvert Cliffs as of 6/30/2015 by Cladding Material

Chapter 6 Thermal Evaluation

NRC RAI 6-1:

Provide additional information and justification that forced cooling (FC) will be maintained during transport and is reliable for recovery action.

The staff reviewed the response to RAI 6-2b ((ADAMS Accession No. ML14288A127) and concluded that it lacks sufficient detail to verify that FC is always operable during transport or FC would be quickly recoverable in the event of failure. The staff performed an independent analysis of the transfer operation and determined that the peak cladding temperature limit of 752°F would be exceeded in the event that FC fails and is not quickly recoverable. Technical Specification (TS) 3.3.2.1 does not currently consider this scenario. Therefore, the staff needs additional information to assure FC will be operable during transport or justify why FC could be quickly recoverable in the event of a failure, or the staff requests that TS 3.3.2.1 be modified to incorporate appropriate actions to be taken in the event that FC cannot be maintained.

This information is required by the staff to determine compliance with 10 CFR 72.128(a)(4).

CCNPP Response to RAI 6-1:

The response was provided in Reference 1.

Chapter 9 Confinement Evaluation

NRC RAI 9-1:

Provide an explanation for the 10% fuel rod rupture assumption during the storage accident condition pressure calculation and confirm the structural integrity of the 32PHB DSC system under accident conditions, assuming 100% fuel rod failure.

Exelon's RAI 9-4 response (ADAMS Accession No. ML14288A127) assumed 10% of the fuel rods fail during the accident blocked vent storage condition. The response also quoted the NUREG-1536 guidance of assuming 100% fuel rod failure during an accident event but provided no justification for not following that guidance.

This information is needed to determine compliance with 10 CFR 72.122.

CCNPP Response to RAI 9-1:

The response was provided in Reference 1.

Chapter 9 Confinement Evaluation

NRC RAI 9-2:

Provide an accurate description of the confinement boundary in the proposed USAR pages in order to provide a complete description of the NUHOMS® 32PHB DSC system.

a) *The confinement boundary is a critical component of the NUHOMS® 32PHB DSC system. As requested in the previously submitted RSI 9-2 (ADAMS Accession No. ML14202A387) and RAI 9-3 (ADAMS Accession No. ML14288A127), a description of the confinement boundary was presented, in text and figure formats. This description should be provided in proposed FSAR pages.*
b) *The RAI 9-3 response sketch of the confinement boundary appears to be different than the actual NUHOMS® 32PHB DSC system, as denoted in drawing NUH32PHB-30-3. The extent of the confinement boundary, especially near the shielding plug assembly and shell, is uncertain and therefore, an updated sketch should be provided.*

This information is needed to determine compliance with 10 CFR 72.24.

CCNPP Response to RAI 9-2:

The response was provided in Reference 1.

Chapter 9 Confinement Evaluation

NRC RAI 9-3:

Provide draft FSAR pages that clearly define that the confinement boundary, which includes the base metal of the components provided in the RSI 9-2 (ADAMS Accession No. ML14202A387) and RAI 9-3 (ADAMS Accession No. ML14288A127) responses and the welds associated with those components, will be helium leak tested to the application's stated 1 E-7 atm cc/sec acceptance criterion of "leaktight", as defined by ANSI N14.5.

As stated in the staff's RAI 1 request, helium leakage testing of the entire confinement boundary to a 1 E-7 ref cc/sec acceptance rate is necessary to demonstrate a "leaktight" system so that release/leakage analyses would be unnecessary. However, parts of the supplied FSAR pages, such as page 13.1-63, indicate that only certain parts of the confinement boundary (e.g., closure welds) are tested to meet a "leaktight" criterion.

This information is needed to determine compliance with 1 0 CFR 72.122.

CCNPP Response to RAI 9-3:

The response was provided in Reference 1.

Chapter 9 Confinement Evaluation

NRC RAI 9-4:

Clarify that the procedures for the helium leak tests in the field, such as during closure, are developed and implemented by an American Society for Nondestructive Testing (ASNT) Level III certified in leakage testing.

Exelon's response to RAI 9-2 (ADAMS Accession No. ML14288A127) indicated that an ASNT non-destructive testing (NDT) Level III procedure would be developed to implement the helium leakage tests during fabrication but did not address the helium leakage tests required to be performed in the field.

This information is needed to determine compliance with 10 CFR 72.122.

CCNPP Response to RAI 9-4:

The response was provided in Reference 1.

Chapter 9 Confinement Evaluation

NRC RAI 9-5:

Confirm that a non-reactive environment will exist within the NUHOMS® 32PHB DSC system during the proposed 40-year license period and specify the fraction and the number of moles of volatile gases, fission gases, and fill gases used in the calculation spreadsheet.

a) The spreadsheet presented in Enclosure 5 of the RAI 9-4 response (ADAMS Accession No. ML14288A127) indicated a helium backfill pressure of 18.2 psia, but there was no analysis to verify that a non-reactive environment would be maintained during the proposed 40-year license period.

b) The previously submitted RAI 9-4 requested justification for the volatile and gas release fractions for the high burnup fuel but the explanation, and corresponding spreadsheet, did not discuss the differences associated with high burnup fuels.

This information is needed to determine compliance with 10 CFR 72.122.

CCNPP Response to RAI 9-5:

The response was provided in Reference 1.

REFERENCE

1. Letter from Mark D. Flaherty (Exelon) to Document Control Desk (NRC), dated June 29, 2015, Response to Second Request for Additional Information Part 1

ENCLOSURE 1

USAR Draft Section 9.6

Calvert Cliffs Nuclear Power Plant
September 11, 2015

9.6 ISFSI LICENSE RENEWAL ACTIVITIES

As part of ISFSI license renewal application an aging management assessment of the HSMs, DSCs [Nutech Horizontal Modular Storage] (NUHOMS-24P ,NUHOMS-32P and NUHOMS-32PHB), transfer cask, transfer cask lifting yoke, the cask support platform, and high burnup fuel was performed. The assessment identified existing activities necessary to provide reasonable assurance that ISFSI and transfer cask components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the renewal period. This section describes these aging management activities that were performed.

9.6.1 AGING MANAGEMENT REVIEW

The purpose of the aging management review is to address the aging effects that could adversely affect the ability of the components and subcomponents to perform their intended function during the period of extended operation. The aging management review involves the following four major steps:

- Identification of in-scope subcomponents requiring aging management reviews
- Identification of materials and environments
- Identification of aging effects requiring management
- Determination of the activities/programs required to manage the effects of aging

The results of this aging management review of the in-scope components are contained in Tables 9.6-1 thru 9.6-6. These tables are the tables developed and submitted in Attachment (3) of Reference 9.21 as part of the license renewal application.

9.6.2 TIME-LIMITED AGING ANALYSIS

Time-Limited Aging Analyses (TLAA) were conducted to identify and evaluate the effect of time-limited aging in order to demonstrate safe operation of the applicable components over the entire period of extended operation. This section discusses the results for each of the TLAAs evaluated for license renewal. The evaluations have demonstrated that the analyses have been projected to the end of the renewed license period.

9.6.2.1 DSC Time-Limited Aging Analysis

AREVA Technical Report 10955-0101 (Reference 9.16) was prepared to identify and evaluate the effect of TLAA, to demonstrate safe operation over the extended service life of the ISFSI For the 24P and 32P DSC systems. The report evaluated the DSCs stored in the HSMs. This section describes the finding of the TLAA.

AREVA Technical Report 10955-TLAA02 (Reference 9.49) was prepared to identify and evaluate the effect of TLAA, to demonstrate safe operation over the extended service life of the ISFSI For the 32PHB DSC systems. The report evaluated the DSCs stored in the HSM-HBs. This section describes the finding of the TLAA.

DSC Materials:

Stainless Steel-DSC Support Rail Steel and DSC Fatigue Evaluation

The NUHOMS-24P ,NUHOMS-32P and 32PHB DSC construction uses Stainless Steel 304 material and compatible weld metal. Since the DSC is filled with the inert helium gas there is no significant corrosion of the DSC shell and other components. Neutron fluence can affect mechanical properties of steels. However, studies on fast neutron damage in stainless steel and low alloy steels rarely evaluate damage at fluence levels below 10^{17} neutrons/cm^2 because they are not significant (Reference 9.17). For the DSC, the neutron fluence (10^{14} neutrons/cm^2) is much less than this level for the intended storage period and hence, a TLAA is not required.

The DSC and weld stresses due to temperature and pressure inside the DSC are an important aspect of the design. Per Reference 9.22, the NUHOMS-24P DSC normal operating design temperature and pressure are 400°F and 10 psig and the accident temperature and pressure are 460°F and 50 psig.

Per Reference 9.16, the NUHOMS-32P DSC normal operating design temperature is 460°F for the DSC shell and 380°F for the welds, top and bottom shield plug/cover plate assembly and the design pressure is 30 psig. The design temperatures for the accident conditions for the NUHOMS-32P DSC shell is 575°F and 475°F for the welds, top and bottom shield plug/cover plate assembly and the design pressure is 100 psig (ref. 9.23). The accident pressure values are established using 100% cladding breach (Ref. 9.24).

Per Reference 9.49, the NUHOMS-32PHB DSC normal operating design temperature is 460°F for the DSC shell and 380°F for the welds, top and bottom shield plug/cover plate assembly and the design pressure is 30 psig. The design temperatures for the accident conditions for the NUHOMS-32PHB DSC shell is 656°F and 610°F for the welds, top and bottom shield plug/cover plate assembly and the design pressure is 100 psig (ref. 9.42 NUH32PHB-0204). The accident pressure values are established using 100% cladding breach (Ref. 9.44 NUH32PHB-0404).

The design temperatures are calculated at the beginning of NUHOMS storage. This calculation is bounding since the DSC temperatures are shown to monotonically decrease as a function of time.

Therefore, the heating effect (and hence, the internal pressure effect) on the DSC for the future 40 years of service will be much less severe than that during the past 20 years of service. Hence, the stresses in the DSC components will be acceptable for the extension period.

The service life for the NUHOMS-24P DSC system is documented in Table 1.2-2, Reference 9.35 as 50 years. Sufficient clearances are provided in both the radial and axial direction between the DSC internal components to permit free thermal expansion for NUHOMS-24P and NUHOMS-32P DSCs and NUHOMS-32PHB DSCs. This design feature acts to minimize the thermal cycling and fatigue on the DSC. There will be

more room for free thermal expansion as the decay heat from the fuel decreases causing the DSC internal component temperatures to decrease as the storage time is increased from 50 years to 60 years. Therefore, thermal cycling and fatigue on the DSC will not be impacted when the storage period is increased from 50 years to 60 years.

Lead in the Shield Plugs

The DSC uses lead in the shield plugs. The lead shielding function is not affected by the radiation level in the DSC.

Poison Plates (NUHOMS-32P and 32PHB DSC Only)

Since the NUHOMS-32P and 32PHB DSCs use fixed neutron absorbers in the DSC basket, calculations were performed to assure criticality safety. Time dependency for criticality calculations may result due to depletion of boron in the poison plates utilized in the NUHOMS-32P and 32PHB DSC baskets.

Reference 9.19, Section 6.3, states that "The continued efficacy of the neutron absorber materials over a 20-year storage period is assured by the design of the NUHOMS-32P and 32PHB canister which ensures that the neutron absorbers will remain in place during accident conditions. Additionally, the neutron flux from the irradiated fuel will result in negligible depletion of the Boron 10 content in the neutron absorber materials over the life of the storage system."

Per Reference 9.16, the total neutron activity in the NUHOMS-32P DSC is 4.175×10^8 n/s/assembly (Ref. 9.25). To estimate the total flux a conservative final assembly surface area of 25,000 cm^2 is considered in Reference 9.16. The total scalar flux is estimated to be 8.65×10^4 n/cm^2-s. Using the thermal cross section for Boron 10, (3,837 barn), the fraction of the original Boron 10 depleted after 60 years is 2.3E-6, which is negligible. The actual neutron flux is mostly fast and epithermal, and will be declining with time, so the actual depletion during dry storage will be less than the depletion calculated in Reference 9.16. Therefore, continued efficacy of the neutron poison is assured for an additional 40 years of fuel storage.

Per Reference 9.49, the total neutron activity in the NUHOMS-32PHB DSC is 4.175×10^8 n/s/assembly.

DSC Fatigue Evaluation

The fatigue evaluation of the NUHOMS-24P DSC is documented in Appendix C.4.1 of Reference 9.16 for a 50 year operational life of the ISFSI. The fatigue effects on the DSC were addressed using the criteria contained in Section III NB-3222.4 of the American Society of Mechanical Engineers (ASME) Code (Reference 9.20, The 1983 Edition). The analysis evaluated the DSC under six criteria and concluded that the DSC and other components satisfy these criteria and that no consideration of fatigue is required for a service life of 50 years.

In order to extend the operational life by another 10 years, it is necessary to re-evaluate the DSC against these six criteria using an approach that is consistent with that utilized in Reference 9.35. A fatigue analysis for the NUHOMS-24P,NUHOMS-32P and 32PHBDSCs was performed for a 60 year service life. This analysis uses the six criteria in NB-3222.4(d) of ASME Code (Reference 9.20, the 1998 Edition including the 1999 Addenda) and determines that the DSC service loads of the NUHOMS-24P DSC ,NUHOMS-32P and NUHOMS-32PHB DSC systems do not create any potential risk of the DSC design fatigue failure and that no detailed fatigue evaluation is necessary (Ref. 9.26).

Fuel Cladding Thickness and Maximum Temperature

The NRC SER (Ref. 9.37) for the Baltimore Gas and Electric Company's SAR for an ISFSI at Calvert Cliffs, Section 2.2.6, Thermal Evaluation, states that "The applicant uses the same methods as in the NUHOMS-24P topical report (Ref. 9.35) for calculating temperature limits for the safe dry storage of spent fuel which resulted in a maximum temperature limit of 335°C for normal operating conditions and 570°C for abnormal and accident conditions."

Further, in order to ensure that the cladding temperature limits are acceptable, three limitations were placed on fuel assemblies that are considered for storage in the NUHOMS®-24P DSC:

1) Maximum burnup and associated maximum decay time of either: of 50,000 MWD/MTU and 12 years; 45,000 MWD/MTU and 14 years; or 42,000 MWD/MTU and 15 years.

2) Maximum initial fill gas pressure less than 435 psia and

3) Maximum individual fuel assembly decay heat of no more than 660 Watts.

The SER further states that "The 335°C limit is conservative when compared to the 340°C temperature limit in the NUHOMS 24P topical report. The 570°C accident temperature limit is also conservative when compared to the measured inert gas (e.g. helium) environment fuel rod failure temperatures of from 765° to 800°C."

The technical basis report for cladding temperature limits, in its concluding section states that "The CSFM model produces a family of generic temperature limit curves to accommodate the variations due to spent fuel storage parameters. These limit curves establish the recommended temperature limits for 40-year dry storage of zircaloy-clad spent fuel." The report also states that "The CSFM methodology has been verified to be technically sound and defensible, and it has a demonstrable degree of conservatism."

The maximum fuel cladding temperature at the initiation of storage for the 24P DSC was calculated to be 322°C (612°F) (Section 2.2.6.2 of Ref. 9.37) for long-term storage ambient temperature of 70°F.

The maximum fuel cladding temperature at the initiation of storage for the 32P DSC was calculated to be 313°C (596°F) (Ref. 9.39 Table 1) and USAR Section 12.8.1.3.2, for long-term storage ambient temperature of 70°F.

The maximum fuel cladding temperature at the initiation of storage for the 32PHB DSC was calculated to be less than 384°C (724°F) (Ref. 9.43 Table 4.3) and USAR Section 13.8.1.3.2, for long-term storage ambient temperature of 70°F.

After 20 years of storage the peak cladding temperatures in both the 24P DSC and 32P DSC and 32PHB DSC are expected to be much lower (Figure 8.1-28 of reference 9.35) due to the monotonic reduction in decay heat as a function of time. Therefore, the fuel cladding temperatures will remain below the limits for the additional 40 years of storage.
The PNL-6189 report (Ref. 9.36) which is used for calculating cladding temperature limit is applicable to 40 years storage and the methodology does not provide for extension to 60 years of storage. The current guidance from NRC on acceptable fuel cladding temperature limits is given in ISG-11, Rev 3 (Ref. 9.38) and no limit on storage term is stated.
The maximum cladding temperature from the PNL-6189 method (635°F, Ref. 9.36) is well below the ISG-11 limit of 752°F.
Hence no further analysis of the cladding thickness or maximum temperature is necessary.

9.6.2.2 HSM and HSM-HB Time-Limited Aging Analysis

Reference 9.16 was prepared to identify and evaluate the effect of HSMs TLAA, to demonstrate safe operation over the extended service life of the ISFSI.
Reference 9.46 and 9.49 were prepared to identify and evaluate the effect of HSM-HBs TLAA, to demonstrate safe operation over the extended service life of the ISFSI.
This section describes the findings of these TLAAs..

HSM and HSM-HB Concrete-Radiation Effects

The HSM is a reinforced concrete structure. The effect of radiation on the HSM concrete is evaluated in USAR Section 8.1.1.5.D ,USAR Section 12.8.1.1.5.D and USAR Section 13.8.1.1.5.D (Ref. 9.45). These evaluations demonstrate that the magnitude of the neutron fluence incident on the concrete is low enough to not affect the properties of the concrete. These evaluations also demonstrate that the magnitude of the gamma-ray energy deposition on the HSM concrete is not sufficient to cause any radiation heating in the concrete of the HSM. Therefore, the thermal analyses documented in Sections 8.1.1.5D and 12.8.1.1.5D and USAR Section 13.8.1.1.5.D (Ref. 9.45) implicitly considered the radiation heat effects adequately for the HSM concrete.

NUHOMS-24P DSC –Decay Heat Effects on HSM

The maximum predicted temperature of concrete at the beginning of storage was estimated to be below 150°F in Reference 9.28 using a bounding decay heat at the beginning of storage life.

The maximum concrete temperatures for the additional 40 years of service (as shown in Figure 8.1-27 of Reference 9.35) will be lower because the

decay heat reduces monotonically as a function of time. Hence, the heating effect on the HSM concrete, for an additional 40 years of service, will be much less severe than the past 20 years of service.

NUHOMS-32P DSC –Decay Heat Effects on HSM

The maximum predicted temperature of the concrete at beginning of storage with a NUHOMS-32P DSC is 157°F per Reference 9.29.

NUHOMS-32PHB DSC –Decay Heat Effects on HSM-HB

The maximum predicted temperature of the concrete at beginning of storage with a NUHOMS-32PHB DSC is 290°F per Reference 9.46.
Note: 32P was also stored in the first four HSM-HB modules. The heat of the 32P is enveloped by the 32PHB heat.

The environmental degradation of reinforced concrete will not be significant, as proper concrete cover has been provided to the reinforcing bars made of carbon steel.

Thermal Cycling Effects on HSM

Reference 9.35, Section 8.2.10.6 documents the analysis of thermal cycling of the HSM based on the 50 year storage life. The number of cycles will increase from 18,250 to 21,900 when the design life is extended from 50 years of storage to 60 years of storage. These are still significantly below the limit of 10,000,000 (See Section 8.2.10.6 of Reference 9.35). Therefore, thermal cycling will have negligible impact on the HSM reinforced concrete for an additional 40 years of service.

Thermal Cycling Effects on HSM-HB

Reference 9.46 documents the HSM-H thermal cycling. Degradation due to elevated temperature is not an aging effect requiring management for the HSM concrete. All of the concrete components of the HSM-HB will not be impaired by thermal cyclic effect or elevated temperature and will be functionally adequate for a total service life of 60 years.

DSC Support Rail Steel in the HSMs

The DSC support structure inside the HSM is designed to support the DSC during normal, off-normal and accident conditions. Since the DSC support rails are fabricated from Nitronic 60 austenitic stainless steel, it is expected that there would be no corrosion of the rail material and is expected to maintain its function for the additional 40 years of service.

DSC Support Rail Lubricant in the HSMs

The HSM and transfer cask support rails are coated with a dry film lubricant Perma-Slik to minimize friction during insertion and retrieval of the DSC.

The material specification of the lubricant indicates that it is suitable for very high and cryogenic temperature applications. The presence of a non-corrosive environment due to the absence of a formal sea breeze and relatively milder temperature fluctuations at ISFSI site ensure that the lubricant does not degrade with age. The effect of radiation on these lubricants is not specified, however, it is expected that it is minimal since these are inorganic and consist entirely of graphite, a moderating material. As stated above once the DSC is in place within the HSM, the lubricant performs no function during storage of the DSC.

NUHOMS-24P DSC –DSC Support Structural Analysis

The coefficient of friction associated with these lubricants is below 0.05 while the design basis calculations employed a coefficient of friction of 0.25 (Section 8.1.1.1 D of Reference 9.35). The mechanical system to be used for DSC transfer is capable of exerting a force equal to the loaded weight of a DSC and this condition has been evaluated in Section 8.1.2.1 of Reference 9.35 for the NUHOMS-24P DSC. A coefficient of friction of 1.0 has been used (for these "jammed DSC" analyses) without relying on the solid film lubricant. The support structure is designed for this loading. Hence, no further analysis is required.

NUHOMS-32P DSC –DSC Support Structural Analysis

The coefficient of friction associated with these lubricants is below 0.05 while the design basis calculations employed a coefficient of friction of 0.25 (Reference 9.16 and 9.23). The mechanical system to be used for DSC transfer is capable of exerting a force equal to the loaded weight of a DSC and this condition has been evaluated for the NUHOMS-32P DSC (Ref. 9.23). A coefficient of friction of 1.0 has been used (for these "jammed DSC" analyses) without relying on the solid film lubricant. The support structure is designed for this loading. Hence, no further analysis is required.

NUHOMS-32PHB DSC –DSC Support Structural Analysis

The coefficient of friction associated with these lubricants is below 0.05 while the design basis calculations employed a coefficient of friction of 0.25 (Reference 9.16 and Reference 9.42 NUH32PHB-0204). The mechanical system to be used for DSC transfer is capable of exerting a force equal to the loaded weight of a DSC and this condition has been evaluated for the NUHOMS-32PHB DSC (Reference 9.42 NUH32PHB-0204). A coefficient of friction of 1.0 has been used (for these "jammed DSC" analyses) without relying on the solid film lubricant. The support structure is designed for this loading. Hence, no further analysis is required.

9.6.2.3 Transfer Cask Fatigue Evaluation

The fatigue evaluation for the transfer cask (Ref. 9.30) is performed in accordance with ASME code criteria listed in Section NC-3219.2 to determine whether the transfer cask service loads of NUHOMS-24P DSC ,NUHOMS-32P and NUHOMS-32PHB DSC systems create potential risk of the design fatigue failure. The criteria evaluation (Ref. 9.30) shows that

transfer cask service loads of NUHOMS-24P and NUHOMS-32P DSC systems do not create potential risk of the transfer cask design fatigue failure, for 600 cycles in 60-year transfer cask life, and that detailed fatigue evaluation is not necessary.

Transfer Cask Trunnions Fatigue Evaluation

The fatigue evaluation of transfer cask trunnions (Ref. 9.31) shows that the transfer cask operations do not pose potential risk of fatigue failure of trunnion or trunnion sleeve throughout the planned 60-year service time.

NS-3 in Transfer Cask

The transfer cask contains 3 inches of NS-3 neutron shielding sandwiched between the cask outer shell material and neutron shield jacket. Per Reference 9.32, the gamma and neutron dose at 1 inch from the cask surface for the accident conditions is 135 mrem/hr and 1000 mrem/hr, respectively for the NUHOMS-24P DSC and 85 mrem/hr and 1433 mrem/hr, respectively for the NUHOMS-32P DSC with burnup limit ≤ 47,000 MWD/MTU and 79 mrem/hr and 1673 mrem/hr, respectively for the 32P DSC with extended burnup limit of ≤ 52,000 MWD/MTU (Ref. 9.25 Tables 6-1 and 6-4).

Per Reference 9.47 NUH32PHB-0502 the gamma and neutron dose at 1 inch from the cask surface for the accident conditions is xx mrem/hr and xxxx mrem/hr, respectively for the NUHOMS-32PHB DSC with burnup limit ≤ 62,500 MWD/MTU.
Per Reference 9.49 NUH32PHB-0500........

Also, per Ref. 9.32 the dose rates at 1 inch from the transfer cask surface for the accident conditions with the NS-3 at the side of the transfer cask replaced with air bounds the dose rates at the inner surface of NS-3 in the transfer cask during normal conditions and that the transfer cask is only subjected to this gamma exposure when a fuel-loaded DSC is in the transfer cask during loading and transfer operations which are short term durations. This results in a gamma dose of approximately 3.0×10^5 Rads over the service life of 60 years. This is based on an assumption that 1 Rad = 1 Rem (Ref. 9.40) and is considered reasonable for gamma radiation for hydrogenous materials. This is significantly below the exposure limit of 1.5×10^{10} Rads for the material as stated in Reference 9.33.

Per Reference 9.16, to estimate the neutron fluence, a neutron dose to flux factor of 1 mrem/hr = 100 n/cm^2-s is used (Ref. 9.34). The dose to flux factor for neutrons is based on dose rate spectra results from various NUHOMS ISFSI evaluations. The integrated fluence is estimated to be approximately 3.16×10^{14} neutrons/cm^2 over the service life of 60 years for the NS-3 in the transfer cask. Reference 9.33 noted that the thermal neutron exposure limit 1.5×10^{19} neutrons/cm^2 for the NS-3 material. Therefore, it is concluded that there is no significant degradation to the NS-3 material for the additional 40 years of operations of the transfer cask.

The exposure to radiation sources for an additional 40 years of service is shown to have no significant impact on the shielding capability of the NS-3 in the transfer cask. No significant hydrogen loss in the NS-3 material is expected due to radiation exposure.

9.6.2.4 Time-Limited Aging Analysis of the Transfer Cask Lifting Yoke

Lifting Yoke Fatigue Evaluation

The fatigue evaluation of the lifting yoke system (Ref.9.31) shows that the transfer cask operations do not pose potential risk of fatigue failure of trunnion or trunnion sleeve throughout the planned 60-year service time.

In the case of the transfer cask lifting yoke assembly, the structural adequacy against fatigue failure is secured for up to 286 transfer cask loading/unloading operations in the planned 60-year service time.

9.6.3 AGING MANAGEMENT PROGRAMS

Aging management programs are developed to identify the activities to be implemented to address the possible aging effects such that no aging effect results in the loss of intended function of the in-scope components. An aging management program is considered effective if it meets one of the following conditions:

- Provides for timely discovery of the effects of aging to be managed
- Mitigates the effects of aging to be managed

Calvert Cliffs will implement and maintain the aging management programs submitted in Attachment 2 of Reference 9.21 for the duration of the ISFSI renewed license operating period.

9.6.3.1 HSM and HSM-HB Aging Management Program

The HSM aging management program credits the Calvert Cliffs Part 50 programs credited for managing the effects of aging in the Calvert Cliffs Nuclear Power Plant as described in Chapter 16 of Reference 9.1. The HSM aging management program involves monitoring the exterior surfaces of the HSMs, including visual inspection of the accessible concrete; any exposed steel subcomponents, embedments, and attachments; and the lightning protection system. Interior inspections are conducted upon loading of a cask. Exterior inspections are conducted annually.

The HSM monitored conditions include, but not limited, to the following:

- Concrete - spalling, cracking, delaminations, honey combs, leaching, discoloration, loss of material, or any other property that would be noted by visual inspection
- Structural Steel - corrosion, peeling paint, deflection, lost or missing anchors/fasteners, missing or degraded grout under base plates, twisted beams, cracked welds
- Equipment Foundations - settlement, cracked concrete
- Equipment Supports - cracked concrete, loose connections, corroded steel

- Roof Systems - structural integrity, deteriorated penetrations (i.e., drains, vents, etc.), signs of water infiltration, cracks, ponding, and flashing degradation
- Seismic Gaps - gaps or loss of joint filler material
- Lightning Protection System (above grade) - corrosion

9.6.3.2 Transfer Cask Aging Management Program

The transfer cask aging management program credits periodic inspections performed on the transfer cask. The procedure includes visual and penetrant test inspections of the carbon steel subcomponents. Monitored aging effects by the aging management program include loss of material due to various forms of corrosion.

9.6.3.2a Forced Cooling Option

No aging management program is required for the forced cooling components. This includes the forced cooling Top Cover Assembly which is Class B and the Wedges at the Bottom of the Cask which are Class C. Both items are stainless steel and are stored in a sheltered environment. Refer to Table 9.6-4 for classifications.
The wedges are rinsed with demineralized water after removal of the cask from the pool and prior to storage per procedure. The Top Cover is never placed in the pool.

9.6.3.3 Transfer Cask Lifting Yoke Aging Management Program

The transfer cask lifting yoke aging management program credits the transfer cask lifting yoke annual inspection. This procedure includes visual and magnetic particle test inspections of the transfer cask lifting yoke carbon steel subcomponents. Monitored aging effects by the aging management program include loss of material due to various forms of corrosion and cracking of material due to stress/strain from lifting.

9.6.3.4 Cask Support Platform Aging Management Program

The cask support platform aging management program credits the Calvert Cliffs power plant Chemistry Control Program, as described in Chapter 16 of Reference 9.1. Loss of material and cracking are prevented through control of specified limits on chloride in the spent fuel pool water.

9.6.3.5 DSC External Surfaces Aging Management Program

The DSC external surfaces AMP consists of condition monitoring activities to confirm there is no degradation of the DSC shell or cover plates that would result in a loss of the pressure/confinement boundary function.

9.6.3.6 High Burnup Fuel Aging Management Program

The high burnup fuel AMP will rely upon joint Electric Power Research Institute and Department of Energy research program as a surrogate program to monitor the condition of high burnup spent fuel assemblies. The results from the research programs will be relied upon to provide reasonable assurance that the high burnup fuel assemblies at Calvert Cliffs

would not experience degradation that would result in a loss of intended functions.

TABLE 9.6-1
AGING MANAGEMENT REVIEW RESULTS FOR THE IRRADIATED FUEL ASSEMBLIES

Subcomponent	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activities
Fuel Rod (Cladding and End Caps)	CC, HT, PB, SS	Zircaloy-4, Zirlo, M5[4]	Air and Gas[1,2] Residual Boron Coating[3]	None Identified	None Required[4]
Guide Tubes	SS	Zircaloy-4 Stainless Steel (Chrome Plated)	Air and Gas[1]	None Identified	None Required[4]
Spacer Grid Assemblies	CC, SS	Zircaloy-4	Air and Gas[1]	None Identified	None Required[4]
Lower End Fitting (and Connectors)	SS (CC, SS)	Stainless Steel Inconel	Air and Gas[1]	None Identified	None Required[4]
Upper End Fitting (Connectors and Holddown Spring)	SS	Stainless Steel Inconel X-750	Air and Gas[1]	None Identified	None Required[4]
Holddown Spring Retainer and Upper End Plugs	None	N/A	N/A	N/A	N/A
Fuel Assembly Control Components	None	N/A	N/A	N/A	N/A
Fuel Rod Pellets and Other Internal Portions	None	N/A	N/A	N/A	N/A

Notes:

1 Air and gas environment outside the fuel rods (inside the DSC) is helium at atmospheric pressure with trace amounts of air and water vapor. Minimal amounts of fission product gases may also be present. Temperature and radiation have been considered as described in Reference 9.41 Section 3.2.3, Environments for the Irradiated Fuel Assemblies.

2 Air and gas environment inside the fuel rods is pressurized helium and fission product gases. Temperature and radiation have been considered as described in Reference 9.41 Section 3.2.3, Environments for the Irradiated Fuel Assemblies.

TABLE 9.6-1
AGING MANAGEMENT REVIEW RESULTS FOR THE IRRADIATED FUEL ASSEMBLIES

[3] Residual boron may coat the irradiated fuel assemblies surfaces since they were exposed to a borated water environment in the SFP prior to storage. Any boric acid residue remaining on the irradiated fuel assemblies will have no deleterious effects due to the minimal amount of water remaining on the irradiated fuel assemblies and the materials of construction for the irradiated fuel assemblies.

[4] A confirmatory program for high burnup fuel is being performed by the Department of Energy which will monitor the condition of high burnup fuel assemblies in dry storage. This confirmatory program includes Zircaloy-4, Zirlo, and M5 cladding materials that have been used at Calvert Cliffs but may not yet have been placed into dry storage at the time of ISFSI license Renewal.

CC Provides criticality control of spent fuel
HT Provides heat transfer
PB Directly or indirectly maintains a pressure boundary (confinement)
SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

TABLE 9.6-2
AGING MANAGEMENT REVIEW RESULTS FOR THE DSCs (NUHOMS-24P AND 32P and 32PHB)

Subcomponent[1]	Intended Function	Materials	Environment	Aging Effect/Mechanism	Aging Management Activities
Basket	SS, CC	Stainless Steel	Air and Gas[2,4]	None Identified	None Required
Guide Sleeves (Basket)	SS, CC	Stainless Steel	Air and Gas[2,4]	None Identified	None Required
Spacer Disks (Basket)	SS,CC	Stainless Steel / Aluminum Coated Carbon Steel	Air and Gas[2,4]	None Identified	None Required
Support Rods (Basket)	SS, CC	Stainless Steel / Aluminum Coated Carbon Steel	Air and Gas[2,4]	None Identified	None Required
Rails (Basket)	SS	Stainless Steel	Air and Gas[2,4]	None Identified	None Required
Rail Inserts (Basket)	SS	Aluminum	Air and Gas[2,4]	None Identified	None Required
Fixed Neutron Absorbers (Basket)	CC	Borated Aluminum Alloy	Sheltered[3]	None Identified	None Required
DSC Shell w/ Bottom shield Plug	PB, SH, SS, HT	Stainless Steel and Lead	Air and Gas[2] Sheltered[3]	Loss of Material due to Crevice and Pitting Corrosion Cracking from Stress Corrosion Cracking	DSC External Surfaces Aging Management Program
Top Shield Plug	PB, SH, SS, HT	Stainless Steel and Lead	Air and Gas[2,4] Sheltered[3]	None Identified	None Required
Cover Plates (Top and Bottom)	PB, SH, SS, HT	Stainless Steel	Sheltered[3]	Loss of Material due to Crevice and Pitting Corrosion Cracking from Stress Corrosion Cracking	DSC External Surfaces Aging Management Program
Siphon and Vent Ports	PB, SH	Stainless Steel	Air and Gas[2,4]	None Identified	None Required
Ram Grapple Ring	SS	Stainless Steel	Sheltered[3]	Loss of Material due to Crevice and Pitting Corrosion Cracking from Stress Corrosion Cracking	DSC External Surfaces Aging Management Program
Dry Film Lubricant	none	N/A	N/A	N/A	none
Swagelok Quick Disconnects	none	N/A	N/A	N/A	none

TABLE 9.6-2
AGING MANAGEMENT REVIEW RESULTS FOR THE DSCs (NUHOMS-24P AND 32P and 32PHB)

Subcomponent[1]	Intended Function	Materials	Environment	Aging Effect/Mechanism	Aging Management Activities
Siphon Tube	none	N/A	N/A	N/A	none
Aluminum coating (Carbon Steel Spacer Discs and Top Shield Plug)	none	N/A	N/A	N/A	none
Nickel Based Thread Lubricant; thread tape or sealant	none	N/A	N/A	N/A	none
Stainless steel plugs/bolts (non-Structural)	none	N/A	N/A	N/A	none
DSC Lifting Lugs	none	N/A	N/A	N/A	None

Notes:

1 Each individual DSC may not contain all of the listed subcomponents.

2 Air and gas environment is helium inside DSC cavity, with possible trace amounts of air, water vapor and fission product gases. Temperature and radiation have been considered as described in Reference 9.41 Section 3.3.3, Environments for the DSCs.

3 Sheltered environment for DSC interior/exterior surfaces that are not part of helium filled DSC cavity.

4 One time short exposure to borated water during loading operations is not considered an environment that impacts long term aging management.

CC Provides criticality control of spent fuel
HT Provides heat transfer
PB Directly or indirectly maintains a pressure boundary (confinement)
SH Provides radiation shielding
SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

TABLE 9.6-3
AGING MANAGEMENT REVIEW RESULTS FOR THE HSM and HSM-HB

Subcomponent	Intended Function	Materials	Environment	Aging Effect/Mechanism	Aging Management Activities
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Cracking due to freeze-thaw degradation	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Loss of material (spalling, scaling) due to freeze-thaw degradation	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Cracking due to moisture, chemical attack, or leaching	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Loss of material (spalling, scaling) due to moisture, chemical attack, or leaching	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	SS	Embedded Steel	Yard, Air	Corrosion due to moisture, chemical attack, or leaching	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Increase in porosity/permeability due to leaching of $Ca(OH)_2$	HSM Aging Management Program

TABLE 9.6-3
AGING MANAGEMENT REVIEW RESULTS FOR THE HSM and HSM-HB

Subcomponent	Intended Function	Materials	Environment	Aging Effect/Mechanism	Aging Management Activities
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Loss of strength due to leaching of $Ca(OH)_2$	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Increase in porosity/permeability due to cement aggregate reactions	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Loss of strength due to cement aggregate reactions	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Cracking due to settlement or loss of bond with embedded steel	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Cracking due to irradiation	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Loss of material (spalling, scaling) due to irradiation	HSM Aging Management Program

TABLE 9.6-3
AGING MANAGEMENT REVIEW RESULTS FOR THE HSM and HSM-HB

Subcomponent	Intended Function	Materials	Environment	Aging Effect/Mechanism	Aging Management Activities
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Cracking due to cement aggregate reaction	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation; Shielded Ventilation Air Inlet Plenum; Ventilation Air Outlet Shielding Blocks	HT, SH, SS	Concrete	Yard, Air	Loss of material due to cement aggregate reaction	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation	HT, SH, SS	Concrete	Embedded/ Underground	Cracking due to freeze-thaw degradation	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation	HT, SH, SS	Concrete	Embedded/ Underground	Loss of material (spalling, scaling) due to freeze-thaw degradation	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation	HT, SH, SS	Concrete	Embedded/ Underground	Cracking due to moisture, chemical attack, or leaching	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation	HT, SH, SS	Concrete	Embedded/ Underground	Loss of material due to moisture, chemical attack, or leaching	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation	SS	Embedded Steel	Embedded/ Underground	Corrosion due to moisture, chemical attack, or leaching	HSM Aging Management Program

TABLE 9.6-3
AGING MANAGEMENT REVIEW RESULTS FOR THE HSM and HSM-HB

Subcomponent	Intended Function	Materials	Environment	Aging Effect/Mechanism	Aging Management Activities
Reinforced Concrete Walls, Roof and Foundation	HT, SH, SS	Concrete	Embedded/ Underground	Increase in porosity/permeability due to leaching of $Ca(OH)_2$	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation	HT, SH, SS	Concrete	Embedded/ Underground	Loss of strength due to leaching of $Ca(OH)_2$	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation	HT, SH, SS	Concrete	Embedded/ Underground	Cracking due to irradiation	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation	HT, SH, SS	Concrete	Embedded/ Underground	Loss of material (spalling, scaling) due to irradiation	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation	HT, SH, SS	Concrete	Embedded/ Underground	Cracking due to cement aggregate reaction	HSM Aging Management Program
Reinforced Concrete Walls, Roof and Foundation	HT, SH, SS	Concrete	Embedded/ Underground	Loss of material due to cement aggregate reaction	HSM Aging Management Program
DSC Structural Steel Assembly	SS	Carbon Steel	Sheltered	Loss of material due to corrosion	HSM Aging Management Program
DSC Structural Steel Assembly	SS	Nitronic 60 Stainless Steel	Sheltered	None Identified	None Required
DSC Seismic Retainer	SS	Carbon Steel	Sheltered	Loss of material due to corrosion	HSM Aging Management Program

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TABLE 9.6-3
AGING MANAGEMENT REVIEW RESULTS FOR THE HSM and HSM-HB

Subcomponent	Intended Function	Materials	Environment	Aging Effect/Mechanism	Aging Management Activities
Cask Docking Flange and Tie Restraints	SS	Carbon Steel	Sheltered	Loss of material due to corrosion	HSM Aging Management Program
Cask Docking Flange and Tie Restraints	SS	Carbon Steel	Yard	Loss of material due to corrosion	HSM Aging Management Program
Heat Shield	HT	Stainless Steel	Sheltered	None Identified	None Required
Shielded Front Access Door and Door Supports	SH, SS	Concrete	Embedded	None Identified	None Identified
Shielded Front Access Door and Door Supports	SH, SS	Carbon Steel	Yard	Loss of material due to corrosion	HSM Aging Management Program
Ventilation Air Openings (One Inlet/ Two Outlets)	HT	Stainless Steel	Yard	None Identified	None Required
Shielded Ventilation Air Inlet Plenum	HT	Stainless Steel	Yard	None Identified	None Required
Ventilation Air Outlet Shielding Blocks	HT	Stainless Steel	Yard	None Identified	None Required
Lighting Protection System	SS	Copper	Yard	None Identified	None Required
Threaded Fasteners and Expansion Anchors	HT, SS	Stainless Steel	Yard Embedded/ Yard	None Identified	None Required
Handrail	SS	Carbon Steel	Yard	None Identified	None Required

HT Provides heat transfer
SH Provides radiation shielding
SS Provides structural support and/or functional support of important to safety equipment (structural integrity)

TABLE 9.6-4
AGING MANAGEMENT REVIEW RESULTS FOR THE TRANSFER CASK (Including FORCED COOLING Option)

Subcomponent	Intended Function	Materials	Environment[1,2]	Aging Effect/Mechanism	Aging Management Activity
Structural Shell (Cask Body)	SS, HT, SH	Carbon Steel	Embedded/Borated Water	Loss of Material due to General Corrosion Loss of Material due to Pitting or Crevice Corrosion	Transfer Cask Aging Management Program
Bottom Support Ring (Cask Body)	SS	Stainless Steel	Sheltered/Borated Water	None Identified	None Required
Bottom Cover Plate (Cask Body)	SS	Stainless Steel	Sheltered/Borated Water	None Identified	None Required
Top Flange (Cask Body)	SS	Stainless Steel	Sheltered/Borated Water	None Identified	None Required
Inner Shell (Cask Body)	SS,HT, SH	Stainless Steel	Sheltered/Borated Water	None Identified	None Required
Lead (Cask Body)	SS,HT	Lead	Embedded	None Identified	None Required
Rails (Cask Attachments)	SS	Stainless Steel (Nitronic 60)	Sheltered	None Identified	None Required
Upper Trunnions (Cask Attachments)	SS	Stainless Steel	Sheltered/Borated Water	Cracking of material due to stress/strain from lifting	Transfer Cask Aging Management Program
Upper Trunnion Sleeves (Cask Attachments)	SS	Stainless Steel	Embedded/Borated Water	Cracking of material due to stress/strain from lifting	Transfer Cask Aging Management Program
Upper Trunnion Nickel Alloy (Cask Attachments)	SS	Inconel	Sheltered/Borated Water	Cracking of material due to stress/strain from lifting	Transfer Cask Aging Management Program
Upper Trunnion Neutron Shielding (Cask Attachments)	HT, SH	Bisco NS-3	Embedded	None Identified	None Required

TABLE 9.6-4
AGING MANAGEMENT REVIEW RESULTS FOR THE TRANSFER CASK (Including FORCED COOLING Option)

Subcomponent	Intended Function	Materials	Environment[1,2]	Aging Effect/Mechanism	Aging Management Activity
Upper Trunnion Cover Plate (Cask Attachments)	SS, SH	Stainless Steel	Sheltered/Borated Water	Cracking of material due to stress/strain from lifting	Transfer Cask Aging Management Program
Lower Trunnions (Cask Attachments)	SS	Stainless Steel	Sheltered/Borated Water	Cracking of material due to stress/strain from lifting	Transfer Cask Aging Management Program
Lower Trunnion Sleeves (Cask Attachments)	SS	Stainless Steel	Embedded/Borated Water	Cracking of material due to stress/strain from lifting	Transfer Cask Aging Management Program
Lower Trunnion Sleeve Nickel Alloy Weld Overlay (Cask Attachments)	SS	Stainless Steel	Embedded/Borated Water	Cracking of material due to stress/strain from lifting	Transfer Cask Aging Management Program
Lower Trunnion Neutron Shielding (Cask Attachments)	HT,SH	Bisco NS-3	Embedded/Borated Water	None Identified	None Required
Ram Access Penetration Ring (Cask Penetration)	SS	Stainless Steel	Sheltered/Borated Water	None Identified	None Required
Upper and Lower Rings, Outer Shell, Relief Valve Support Plates (Cask Neutron Shield)	SH,HT	Stainless Steel	Sheltered/Borated Water	None Identified	None Required
Inner and Outer Support Angle	SH, HT	Stainless Steel	Sheltered/Borated Water	None Identified	None Required

TABLE 9.6-4
AGING MANAGEMENT REVIEW RESULTS FOR THE TRANSFER CASK (Including FORCED COOLING Option)

Subcomponent	Intended Function	Materials	Environment[1,2]	Aging Effect/Mechanism	Aging Management Activity
(Cask Neutron Shield)					
Shielding Material (Cask Neutron Shield)	SH, HT	Bisco NS-3	Embedded	None Identified	None Required
Inner, Outer, and Side Plates (Top Cover Assembly)	SS	Stainless Steel	Sheltered	None Identified	None Required
Ring; Eye Bolt Stand-offs (Top Cover Assembly)	SS	24 Hot Galvanized Finish	Sheltered	None Identified	None Required
Neutron Shielding (Top Cover Assembly)	SH	Bisco NS-3	Embedded	None Identified	None Required
Inner, Outer, and Side Plates (Bottom Cover Assembly)	SH	Stainless Steel	Sheltered/Borated Water	None Identified	None Required
Bottom Cover O-ring Seals	HT, SH	Polymer (Ethylene Propylene)	Sheltered	Materials Property Change	N/A – subject to routine replacement
Neutron Shielding (Bottom Cover Assembly)	SH	Bisco NS-3	Embedded	None Identified	None Required
Cask Bottom Cover Plate	SH	Stainless Steel	Sheltered	None Identified	None Required
Neutron Shielding (Cask Bottom)	SH	Bisco NS-3	Embedded	None Identified	None Required
Bolts, Washers, and Threaded	SH	Carbon Steel	Sheltered	Loss of Material due to General Corrosion	Transfer Cask Aging Management Program

TABLE 9.6-4
AGING MANAGEMENT REVIEW RESULTS FOR THE TRANSFER CASK (Including FORCED COOLING Option)

Subcomponent	Intended Function	Materials	Environment[1,2]	Aging Effect/Mechanism	Aging Management Activity
Fasteners for Top Cover Plate and Ram Access Plate				Loss of Material due to Pitting or Crevice Corrosion	
Misc Subcomponents	none	N/A	N/A	N/A	N/A
Forced Cooling Subcomponent Top Cover Assembly[3]	HT	Stainless Steel	Sheltered	None Identified	None Required Class B
Forced Cooling Subcomponent Wedges at Bottom of Cask	HT	Stainless Steel	Sheltered	None Identified	None Required Class C

Notes:

1 Sheltered environment represents ambient conditions on the interior of the transfer cask, conservatively including connecting and embedded surfaces. Some subcomponents may have interior surfaces that are considered embedded. No aging effects are identified for the embedded surfaces and no aging management is required. Temperature and radiation were considered, as described in Reference 9.41 Section 3.5.3, Environments for the Transfer Cask.

2 All subcomponents that are immersed in the borated water of the SFP are rinsed off with deionized water after use.

3 The Forced Cooling lid is not immersed in the SFP.

HT Provides heat transfer
SH Provides radiation shielding
SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

TABLE 9.6-5
AGING MANAGEMENT REVIEW RESULTS FOR THE TRANSFER CASK LIFTING YOKE

Subcomponent	Intended Function	Materials	Environment[1]	Aging Effect/Mechanism	Aging Management Activity
Lifting Hook Plates	SS	Carbon Steel	Sheltered/Borated Water	Loss of Material due to General Corrosion Loss of Material due to Pitting or Crevice Corrosion Cracking of material due to stress/strain from lifting	Transfer Cask Lifting Yoke Aging Management Program
Lifting Beam Plates	SS	Carbon Steel	Sheltered/Borated Water	Loss of Material due to General Corrosion Loss of Material due to Pitting or Crevice Corrosion Cracking of material due to stress/strain from lifting	Transfer Cask Lifting Yoke Aging Management Program
Later Brace Plates	SS	Carbon Steel	Sheltered/Borated Water	Loss of Material due to General Corrosion Loss of Material due to Pitting or Crevice Corrosion	Transfer Cask Lifting Yoke Aging Management Program
Support Brace Plates	SS	Carbon Steel	Sheltered/Borated Water	Loss of Material due to General Corrosion Loss of Material due to Pitting or Crevice Corrosion	Transfer Cask Lifting Yoke Aging Management Program
Pin (Round Bar)	SS	Stainless Steel	Sheltered/Borated Water	Loss of Material due to General Corrosion Loss of Material due to Pitting or Crevice Corrosion	Transfer Cask Lifting Yoke Aging Management Program

TABLE 9.6-5
AGING MANAGEMENT REVIEW RESULTS FOR THE TRANSFER CASK LIFTING YOKE

Subcomponent	Intended Function	Materials	Environment[1]	Aging Effect/Mechanism	Aging Management Activity
Pin Handle	None	N/A	N/A	N/A	N/A
Pin Cradle Pipe	None	N/A	N/A	N/A	N/A
Rear Pin Stop	None	N/A	N/A	N/A	N/A
Pin Lock	None	N/A	N/A	N/A	N/A
Main Assembly Bolts, Nuts, Washers	SS	Carbon Steel	Sheltered	Loss of Material due to General Corrosion Loss of Material due to Pitting or Crevice Corrosion	Transfer Cask Lifting Yoke Aging Management Program
Support angles and misc hardware	None	N/A	N/A	N/A	N/A
Hook Bearing Plate	SS	Bronze	Sheltered	None Identified	None Required

Notes:

[1] All subcomponents that are immersed in the borated water of the SFP are rinsed off with deionized water after use.

SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

TABLE 9.6-6
AGING MANAGEMENT REVIEW RESULTS FOR THE CASK SUPPORT PLATFORM

Subcomponent	Intended Function	Materials	Environment	Aging Effect/Mechanism	Aging Management Activity
Base Plate	SS	Stainless Steel	Borated Water	Loss of material due to pitting or stress corrosion cracking	Cask Support Aging Management Program
Web Plates	SS	Stainless Steel	Borated Water	Loss of material due to pitting or stress corrosion cracking	Cask Support Aging Management Program
Mid Plate	SS	Stainless Steel	Borated Water	Loss of material due to pitting or stress corrosion cracking	Cask Support Aging Management Program
Top Plate	SS	Stainless Steel	Borated Water	Loss of material due to pitting or stress corrosion cracking	Cask Support Aging Management Program
Honeycomb Energy Absorber	SS	Aluminum	Compressed Air	None Identified	None Required
Honeycomb Base Plate	SS	Stainless Steel	Borated Water	Loss of material due to pitting or stress corrosion cracking	Cask Support Aging Management Program
Honeycomb Casing Plate	SS	Stainless Steel	Borated Water	Loss of material due to pitting or stress corrosion cracking	Cask Support Aging Management Program
Honeycomb Outer Plate	SS	Stainless Steel	Borated Water	Loss of material due to pitting or stress corrosion cracking	Cask Support Aging Management Program
Bottom Location Plates	SS	Stainless Steel	Borated Water	Loss of material due to pitting or stress corrosion cracking	Cask Support Aging Management Program
Lifting Lugs	SS	Stainless Steel	Borated Water	Loss of material due to pitting or stress corrosion cracking	Cask Support Aging Management Program
Tubing Manifold, Relief Valve, Pressure Gauge, Quick-Connect	none	N/A	N/A	N/A	N/A

SS Provides structural support and/or functional support of important to safety equipment (structural integrity)
N/A Not applicable

9.8 REFERENCES

9.1 Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company

9.2 SFSP Procedure SFSPP-1, Preparation and Control of Spent Fuel Storage Procedures

9.3 SFSP Procedure SFSPP-2, Control of Changes and Deviations Found During Construction

9.4 SFSP Procedure SFSPP-4, Procurement

9.5 SFSP Procedure SFSPP-13, Records Retention

9.6 SFSP Procedure SFSPP-14, Nuclear Related Indoctrination, Training and Qualification

9.7 QAP-36, Independent Spent Fuel Storage Installation

9.8 10 CFR Part 72 Quality Assurance Program for the Spent Fuel Storage Project

9.9 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated December 20, 1990, Response to NRC's Comments on the Safety Analysis Report (SAR) for BGE's License Application for Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI)

9.10 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated September 30, 1991, Response to NRC's Follow Up Comments on the Safety Analysis Report (SAR) for BGE's License Application for Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI)

9.11 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated December 27, 1991, Response to Requests for Additional Information (RAI), Dated December 12 and 19, 1991, on the Safety Analysis Report (SAR) for BGE's License Application for Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI)

9.12 Letter from Charles H. Cruse (BGE) to Region I, Regional Administrator (NRC), dated October 19, 1993, Calvert Cliffs Nuclear Power Plant Independent Spent Fuel Storage Installation; Docket No. 72-8150-317/318, Preoperational Test Acceptance Criteria and Test Results

9.13 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated August 18, 1992, Revision to the ISFSI Decommissioning Plan

9.14 CCNPP Calculation No. CA06354 Rev.0 (Hopper Elmore Associates Calculation HABGE-12/03-1199 Rev.0), "Accidental Drop Loading Evaluation of 14x14 Fuel Assembly with Missing Fuel Rods"

9.15 CCNPP ES200600043, Implementation of ISFSI License Amendment No. 9 Prior to 2010 Loadings

9.16 CCNPP Calculation No. CA07335 Rev.0 (AREVA Technical Report 10955-0101, Revision 1, May 21, 2010), ISFSI Time-Limited Aging Analysis Report

9.17 U.S. Nuclear Regulatory Commission Regulatory Guide 1.99, Revision 2, May 1988, Radiation Embrittlement of Reactor Vessel Materials

9.18 BLANK

9.19 Letter from R. J. Lewis (NRC) to G. Vanderheyden (CCNPP), dated June 10, 2005, Amendment 6 to Material License No. SNM-2505 for the Calvert Cliffs Independent Spent Fuel Storage Installation, Safety Evaluation Report

9.20 American Society of Mechanical Engineers (ASME), Boiler & Pressure Vessel Code, Section III, Division I, Subsection NB, July 1, 2009, Class 1 Components, Rules for Construction of Nuclear Facility Components

9.21 Letter from G. H. Gellrich (Exelon) to Document Control Desk (NRC), dated September 18, 2014, Response to Fourth Request for Additional Information for Renewal Application to Special Nuclear Materials License No. 2505 for the Calvert Cliffs Site Specific Independent Spent Fuel Storage Installation (TAC No. L24475)

9.22 CCNPP Calculation No. CA04132 Rev.3 (Hopper and Associates Calculation HABGE-02/98-0610 Rev.4) NUHOMS® 24P ISFSI DSC Structural Analysis For DSC Assemblies R001 Through R024.

9.23 CCNPP Calculation No. CA06359 (Transnuclear Calculation 1095-34 Rev.5) NUHOMS®32P ISFSI DSC Structural Analysis

9.24 CCNPP Calculation No. CA06300 (Transnuclear Calculation 1095-20 Rev.1) NUHOMS® 32P Maximum Operating Pressure, Storage and Transfer

9.25 CCNPP Calculation No. CA06750 Rev.0 Loading and Transfer Does Rates for ISFSI 32P Burnup Extension

9.26 CCNPP Calculation No. CA07337 (AREVA Technical Report 10955-0202 Rev.1) DSC Fatigue Analysis for NUHOMS® 24P and NUHOMS® 32P

9.28 CCNPP Calculation No. CA03948 Rev.0 (Nutech Calculation BGE001.0402 Rev.0) NUHOMS® HSM Heat Transfer

9.29 CCNPP Calculation No. CA06301 Rev.0 (Transnuclear Calculation 1095-18 Rev.0) HSM Thermal Analysis-Normal Storage Conditions

9.30 CCNPP Calculation No. CA07338 (Transnuclear Calculation, 10955-0203 Rev.1) Transfer Cask Fatigue Analysis for NUHOMS 24P and NUHOMS 32P

9.31 CCNPP Calculation No. CA07336 (Transnuclear Calculation, 10955-0201 Rev.1) Fatigue Analysis for NUHOMS® 24P and NUHOMS® 32P Transfer Cask Trunnion and Lifting Yoke System

9.32 CCNPP Calculation No. CA06292 (Transnuclear Calculation 1095-49 Rev.0) NUHOMS®-32P Radiation Dose Rates Loading and Transfer

9.33 BISCO Products, Inc., "NS-3 Castable Neutron and/or Gamma Shielding Material," specification sheet. Transnuclear File No. DUK003.0016.2.

9.34 CoC 1030, Updated Final Safety Analysis Report for the NUHOMS® HD System for Irradiated Nuclear Spent Fuel, Revision 2.

9.35 Topical Report for the NUTECH® Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS® -24P (NUH002.0100), Rev. 1A, NRC Docket No. M-39.

9.36 Pacific Northwest Laboratory Report PNL-6189/UC-85, "Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas," by Levy I.S. et al, May 1987.

9.37 NRC Safety Evaluation Report for the BG&E's Safety Analysis Report for an Independent Spent Fuel Storage Installation at Calvet Cliffs, November 1992.

9.38 "Cladding Considerations for Transportation and Storage of Spent Fuel", ISG-11 Rev.3, NRC.

9.39 CCNPP Calculation No. CA06312 Rev.0 (Transnuclear Calculation, 1095-56 Rev.0) Thermal Analysis of Storage Cases Poison Plates in the Basket

9.40 Samuel Glasstone and Alexander Sesonske, "Nuclear Reactor Engineering," Third Edition, Van Nostrand Reinhold Press.

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9.42 CCNPP Calculation No. CA07303 (Transnuclear Calculation NUH32PHB-0204 Rev.0) NUHOMS®32PHB ISFSI DSC Structural Analysis for Storage and Onsite Transfer Loads

9.43 CCNPP Calculation No. CA07320 (Transnuclear Calculation NUH32PHB-0403 Rev.1) NUHOMS® 32PHB Thermal Evaluation for Storage and Transfer Conditions

9.44 CCNPP Calculation No. CA07321 (Transnuclear Calculation NUH32PHB-0404 Rev.0) NUHOMS® 32PHB Internal Pressure for Storage and Transfer Conditions

9.45 CCNPP Calculation No. CAxxxx (AREVA Calculation 10955-0500 Rev.0) NUHOMS®32PHB DSC Poison Plate & TC NS-3 Depletion Evaluation

9.46 CCNPP Calculation No. CA10031 (AREVA Calculation 10955-TLAA01 Rev.0) Time-Limited Aging Analysis (TLAA) of HSM-HB Concrete for Thermal Considerations

9.47 CCNPP Calculation No. CA07329 (Transnuclear Calculation NUH32PHB-0502) NUHOMS® 32PHB Radiation Dose Rates for Loading and Transfer

9.48 CCNPP Calculation No. CA07310 (Transnuclear Calculation NUH32PHB-0211) "Reconciliation for Transfer Cask CCNPP-FC Structural Evaluation".

9.49 CCNPP Calculation No. CAxxxx (AREVA Calculation 10955-TLAA02 Rev.0) Time-Limited Aging Analysis (TLAA) of NUHOMS® 32PHB System