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a joint venture of
Constellation Energy · eDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

June 28, 2011

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Independent Spent Fuel Storage Installation
Material License No. SNM-2505, Docket No. 72-8
Responses to Request for Additional Information, RE: Calvert Cliffs Independent Spent Fuel Storage Installation License Renewal Application

REFERENCES:

(a) Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated September 17, 2010, Site-Specific Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application

(b) Letter from Mr. J. Goshen (NRC) to Mr. G. H. Gellrich (CCNPP), dated April 28, 2011, First 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)

In Reference (a), Calvert Cliffs Nuclear Power Plant, LLC (Calvert Cliffs) submitted Calvert Cliffs site-specific Independent Spent Fuel Storage Installation license renewal application. In Reference (b), the Nuclear Regulatory Commission issued a request for additional information to support their review of Calvert Cliffs site-specific Independent Spent Fuel Storage Installation license renewal application. Attachment (1) contains Calvert Cliffs response to the request for additional information.

NMSS26

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, George H. Gellrich, being duly sworn, state that I am Vice President - Calvert Cliffs Nuclear Power Plant, LLC (CCNPP), and that I am duly authorized to execute and file this response on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of St. Mary's, this 28th day of June, 2011.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires: March 14 2015
Date

GHG/KLG/bjd

Attachments: (1) Calvert Cliffs Response to NRC Request for Additional Information
Enclosures: 1. RAI 3-1 Tables
2. Table 3.4-1 Revision

cc: D. V. Pickett, NRC
W. M. Dean, NRC
Resident Inspector, NRC
S. Gray, DNR
J. Goshen, NMSS
E. Ghigiarelli, MDE
V. Ordaz, NMSS

ATTACHMENT (1)

CALVERT CLIFFS RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

Calvert Cliffs Nuclear Power Plant, LLC
June 28, 2011

CALVERT CLIFFS NUCLEAR POWER PLANT, LLC
SPECIAL NUCLEAR MATERIALS LICENSE NO. 2505
DOCKET NO. 72-8
LICENSE RENEWAL REQUEST
FIRST REQUEST FOR ADDITIONAL INFORMATION

By letter dated September 17, 2010, as supplemented February 10, and March 9, 2011, Calvert Cliffs Nuclear Power Plant (CCNPP), LLC, submitted a license renewal application (LRA) to the U.S. Nuclear Regulatory Commission (NRC) for the CCNPP site-specific Independent Spent Fuel Storage Installation (ISFSI). The NRC staff has reviewed your application and has determined that additional information is required to complete its detailed technical review.

REQUEST FOR ADDITIONAL INFORMATION

ACRONYMS

AMP	*Aging Management Program*
CAP	*Corrective Action Program*
CCNPP	*Calvert Cliffs Nuclear Power Plant Llc*
DSC	*Dry Shielded Canister*
HSM	*Horizontal Storage Module*
ISFSI	*Independent Spent Fuel Storage Installation*
NRC	*U.S. Nuclear Regulatory Commission*
RAI	*Request For Additional Information*
SSC	*Structures, Systems, and Components*
UFSAR	*Update Final Safety Analysis Report*

Chapter 3: Aging Management Reviews

NRC RAI 3-1:

Provide a list identifying all nonquantifiable phrases used in the license renewal application and the revised ISFSI UFSAR and Changes. If applicable, replace the nonquantifiable phrases with statements that can provide justifiable and/or measureable values.

Throughout the license renewal application and in the revised ISFSI UFSAR Supplement and Changes, nonquantifiable phrases were used (e.g., significant, not significant, small, slightly). Please identify each nonquantifiable phrase, and if applicable, replace with statements that can provide justifiable and/or measurable values.

This is required to evaluate compliance with 10 CFR 72.24(c) and (d).

CCNPP Response 3-1:

Calvert Cliffs conducted a global word search on the following qualitative terms that were used in Calvert Cliffs ISFSI License Renewal application (Reference 1) and in the revised ISFSI USAR supplement:

small	large	slight
significant	moderate	low
minor	many	few
little	less	most
sufficient		

Approximately 85 occurrences of these terms were identified. Each occurrence of the aforementioned terms was screened using the following criteria to determine if additional quantitative/descriptive information was needed.

1. Terms "Screens In"

   Meaning: Use of the term requires additional consideration if it is used for one of the following reasons:

   a. The term characterizes an aging effect (e.g., degradation, cracking, fatigue, corrosion, loss of material, change in properties, etc.); or
   b. The term provides important information about the operation, function, or other characteristics of an "In-Scope" SSC per Table 2.3-1 of Calvert Cliffs License Renewal submittal (e.g., HSMs, DSCs, IFAs, Transfer Cask, Cask Lifting Yoke, Cask Support Platform, Spent Fuel Cask Handling Crane); or
   c. The term is used to describe dose, environmental impact, or other hazard such as combustible material, dust.

   If a term screens in, one of the following steps were performed:

   - Provide quantitative information if it is available;
   - Provide additional description; or
   - Define the meaning of the term (e.g., "insignificant means the ability of the "in scope" SSC to provide its safety function is not impaired").

2. Terms "Screens Out"

   Meaning: Use of the term is considered not material to Calvert Cliffs License Renewal submittal for one of the following reasons:

   a. The term is included in the title of reference document;
   b. The term is included in a quote;
   c. The term is explained by adjacent quantitative information [e.g., "small (less than 20 percent)"];
   d. Use of the term is not related to any of the following:
      - "In Scope" SSCs per Table 2.3-1 of Calvert Cliffs License Renewal submittal,
      - Aging effect,
      - Dose, environmental impact, or other hazard (e.g. combustible material);
   e. Use of the term does not provide important information. It is merely descriptive and the meaning of the statement is not changed if the term were deleted. (For example, the word "small" could be deleted from the following statement without altering the meaning "Water in the grapple ring is drained through a small hole.")

   Table 1 of Enclosure 1 provides the results of the screenings performed for Reference 1 where, as a result of the screening process, additional information was determined to be needed (i.e., usages that "Screened In"). Table 2 of Enclosure 1 provides the results that were "Screened Out".

NRC RAI 3-2:

Provide a citation for the calculations supporting the use of a higher burnup limit of 52 GWd/MTU for the NUHOMS-32P DSC as described in Section 3.4.3, Environments for the HSMs.

In Section 3.4.3, the applicant states that "Calculations supporting the use of the higher burnup limit of 52 GWd/MTU for the NUHOMS-32P DSC demonstrated the gamma energy fluence from the higher *burnup NUHOMS-32P DSC would also be less than the gamma energy fluence calculated in the NUHOMS-24P DSC. As a result, there will only be a negligible rise in concrete temperature. The calculation further demonstrated the fluence level will remain well below the threshold value that would cause neutron induced degradation of HSM concrete. However, the applicant did not cite any references for these calculations.*

This is required to evaluate compliance with 10 CFR 72.24(c) and (d).

CCNPP Response 3-2:

The document being referenced is Calvert Cliffs Calculation CA06751 which was previously submitted to the NRC (see ADAMS Accession Number ML091680545). Page 20 of that calculation provides the neutron and gamma flux calculated at various locations inside the HSM using the design basis MCNP model for the NUHOMS-32P DSC. This was accepted by the NRC in Calvert Cliffs ISFSI License Amendment 9.

NRC RAI 3-3:

Confirm whether the copper subcomponents of the lightning protection system in Table 3.4-1 are associated with the bronze subcomponents of the lightning protection system that require aging management. If they are, provide rationale that the copper subcomponents of the lightning protection system exposed to a yard environment are not subject to the aging effect of loss of material due to general corrosion and pitting.

In Section 3.4.5 the applicant identified that the aging effect of loss of material due to general corrosion and pitting could occur for the bronze subcomponents of the lightning protection system exposed to a yard environment. However, no aging effect is identified for the copper subcomponents of the lightning protection system exposed to a yard environment in Table 3.4-1.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response 3-3:

In Reference 1, Section 3.4.5, we incorrectly included bronze subcomponents of the lightning protection system as a metal subcomponent of the HSM that requires management of age related degradation. The only HSM subcomponents that require management for the aging effect of loss of material due to general corrosion and pitting are the carbon steel subcomponents of the HSMs.

Copper and copper alloys are widely used in many environments and applications because of their excellent corrosion resistance, which is coupled with combinations of other desirable properties, such as superior electrical and thermal conductivity, ease of fabricating and joining, wide range of attainable mechanical properties, and resistance to biofouling. Bronze is a copper alloy and thus has excellent corrosion resistance. As a copper alloy, the bronze subcomponents (lightning rods) of the lightning protection system are considered part of the copper material group for the lightning protection system that was evaluated in Reference 1, Table 3.4-1 as not requiring aging management. Enclosure (2) contains a revision to Table 3.4-1 to more clearly show the inclusion of the bronze subcomponents as not requiring aging management due to general corrosion.

NRC RAI 3-4:

Provide rationale that the underground concrete subcomponents of the HSM exposed to a soil or groundwater environment are not subject to the aging effects of cracking, loss of bond, and loss of material due to corrosion.

In Section 3.4.5 the applicant considered the aging effects of loss of material, cracking, and change in material properties that require management for the above-grade concrete subcomponents of the HSM. NUREG-1801, Rev. 2, indicates that the aging effects of cracking, loss of bond, and loss of material could occur for concrete structures exposed to a soil or groundwater environment due to corrosion of the embedded steel rebar.

However, the applicant did not identify any aging effects that require management for the underground concrete structures.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response 3-4:

The aging effects of the underground concrete subcomponents of the HSM exposed to a soil or groundwater environment are monitored through the same inspection credited for the above ground concrete portion of the HSM. On an annual basis Calvert Cliffs conducts a visual inspection of all the ISFSI concrete structures. This inspection looks for areas of spalling and cracking concrete. A corrective action report would then be initiated for areas showing such signs of degradation.

Reference 1, Table 3.4-1 (attached as Enclosure 2 to this document) has been revised to reflect the inclusion of aging effects for underground concrete structures.

Table 1 below contains the Calvert Cliffs site groundwater chemistry results at the ISFSI site taken in May 2011. The groundwater pH is above the threshold limit of where potential concrete degradation may occur. The chloride and sulfate concentrations are also well below threshold concentrations. Therefore, the potential for the concrete below grade within the HSM space to degrade due to chemical attack is considered negligible.

Table 1

Parameter	Threshold Limit When Degradation Occurs*	Calvert Cliffs Groundwater Chemistry
pH	<5.5	6.6
Chlorides	>500 ppm	16 ppm
Sulfates	>1500 ppm	0.96 ppm

* Threshold limits obtained from Reference 2.

NRC RAI 3-5:

Provide rationale that the stainless steel subcomponents of the HSM exposed to a yard or yard-salt environment are not subject to the aging effects of stress corrosion cracking and loss of material due to corrosion.

In Section 3.4.5 the applicant did not identify any aging effects that require management for a number of stainless steel subcomponents exposed to a yard environment. NUREG-1801, Rev. 2, indicates that the aging effects of stress corrosion cracking and loss of material due to pitting and crevice corrosion could occur for stainless steel components exposed to an outdoor air environment. The corrosion testing documented in NUREG/CR-7030 shows that stainless steel materials are susceptible to stress corrosion cracking in coastal marine environments.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response 3-5:

Calvert Cliffs is aware that the susceptibility of the stainless steel dry storage containers (DSC) to stress corrosion cracking when exposed to a marine air environment has recently emerged as a potential issue through the conduct of testing performed in NUREG/CR-7030. However at this point there is clearly no consensus amongst industry and the NRC as to the applicability of the findings of the NUREG to stainless steel DSCs in operation at Calvert Cliffs and throughout the industry. In addition there is no agreed upon threshold for chloride concentration, stress conditions and temperature conditions necessary for stress corrosion cracking to be a concern. Calvert Cliffs and Constellation Energy Nuclear Group are participating with industry (NEI/EPRI/vendor) initiatives evaluating this issue and will follow recommendations and guidance that might come from these initiatives and that are applicable to Calvert Cliffs.

Calvert Cliffs ISFSI is located approximately 2800 feet from the western shoreline of the Chesapeake Bay and is situated at an elevation of 114 feet above mean sea level. At Calvert Cliffs location along the Chesapeake Bay, the bay water is considered brackish water with lower salt content than exists in the ocean. Chesapeake Bay salinity levels in the area of Calvert Cliffs have historically varied seasonally between 5 and 20 parts per thousand. These values are well below the average seawater salinity value of 35 parts per thousand. All of these factors reduce the likelihood that the air environment actually experienced by Calvert Cliffs DSCs, located inside their concrete storage modules, is similar to the air environment simulated in the NUREG testing.

NRC RAI 3-6:

Confirm whether the ISFSI AMP listed in the sixth column of Table 3.4-1 is identical to the HSM APM described in Section A2.1.

In Section 3.4.6 the applicant credited the HSM AMP to manage the aging effects for the carbon steel subcomponents and above-grade concrete structures of the HSM. The sixth column of Table 3.4-1 identified the ISFSI AMP for managing the aging effects.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response 3-6:

Yes, the ISFSI AMP listed in Reference 1, the sixth column of Table 3.4-1, and discussed in Section A 2.1 covers the aging management activities for the HSMs. There is no separate HSM AMP.

The activities associated with the ISFSI aging management program, when continued in the renewed license period, will manage the aging effects for the steel portion of the HSMs, and will include the conservative evaluation of the condition of accessible concrete for the HSM subcomponents identified in Table 3.4-1.

NRC RAI 3-7:

Provide the chemical conditions of the yard-salt environment in terms of pH and chloride concentration.

Table 3.4-1 includes a yard-salt environment to which the concrete subcomponents of the HSM are exposed. No data on the yard-salt environment were provided.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response 3-7:

The environment listed for reinforced concrete walls, roofs and foundations in Reference 1, Table 3.4-1 as being "Yard-Salt" was incorrect. Instead its environment should have been listed simply as "Yard." This would make it consistent with all other references to "Yard" environments in Table 3.4-1. There is no reason to suspect this is an applicable aging mechanism here at Calvert Cliffs. As discussed in our response to RAI 3.5, Calvert Cliffs ISFSI is located approximately 2800 feet from the western shoreline of the Chesapeake Bay and is situated at an elevation of 114 feet above mean sea level. At Calvert Cliffs location along the Chesapeake Bay, the bay water is considered brackish water with lower salt content than exists in the ocean. Chesapeake Bay salinity levels in the area of Calvert Cliffs historically have varied seasonally between 5 and 20 parts per thousand. These values are well below the average seawater salinity value of 35 parts per thousand.

In addition, as also indicated in our response to RAI 3-5, Calvert Cliffs will continue to participate in industry initiatives in this area and will follow industry developed recommendations and guidelines should it be shown that conditions at Calvert Cliffs ISFSI are conducive to this aging mechanism.

NRC RAI 3-8:

Provide copies of applicable preventive maintenance program procedures for the management of fatigue, wear, and mechanical degradation of carbon steel wire rope and general corrosion of carbon steel components of the spent fuel cask handling crane. Describe the inspection requirements and frequencies for the preventive maintenance Program to ensure that the aging effects are adequately managed.

In Section 3.8 the applicant credited the AMPs under 10 CFR Part 50 for managing the aging effects of the spent fuel cask handling crane.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response 3-8:

The age management program for the spent fuel cask handling crane was addressed as part of Calvert Cliffs operating plant License Renewal application (Reference 3) under 10 CFR Part 50. As such no 10 CFR Part 72 aging management review was performed. The aging management review of the spent fuel cask handling crane was discussed in Reference 3, Appendix A, Section 3.2.2. The NRC accepted Calvert Cliffs aging management program in Reference 4, Section 3.11.

While the current operating licenses of the Calvert Cliffs Units will expire (2034 and 2036) before the end of the requested ISFSI license renewal period (2052), even if Calvert Cliffs chooses not to further extend the operating licenses, Calvert Cliffs will remain responsible for maintaining the spent fuel pool and associated support equipment until such time that all irradiated fuel assemblies are transferred offsite. The requirements of 10 CFR 50.54(bb) will ensure components covered under 10 CFR Part 50 which support the site's ISFSI operation will continue to be appropriately maintained for the remaining duration of the ISFSI renewal period. Therefore Calvert Cliffs will commit to maintain the existing 10 CFR Part 50 aging management program for the spent fuel cask handling crane for the duration of the proposed ISFSI License Renewal period.

NRC RAI 3-9:

Provide the environmental conditions to which the carbon steel wire rope is exposed and rationale that the carbon steel wire rope is not subject to the aging effect of general corrosion.

In Section 3.8 the applicant identified the aging effects of fatigue, wear, and mechanical degradation of carbon steel wire rope and general corrosion of carbon steel components of the spent fuel cask handling crane in reference to Table 16-2, Item 11 of the nuclear power plant UFSAR.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response 3-9:

As discussed in our response to RAI 3-8 the age management program for the spent fuel cask handling crane was addressed as part of Reference 3 under 10 CFR Part 50. This aging management review was discussed in Reference 3, Section 3.2.2 and it includes a discussion of the environment and the identification of general corrosion as an aging mechanism for the carbon steel wire rope. This review credited existing practices of conducting an inspection prior to each use and the performance of a preventive maintenance activity to conduct an annual visual inspection. These activities were assessed as being effective methods for the discovery and management of this aging mechanism.

NRC RAI 3-10:

Provide copies of applicable performance evaluation program procedures for the management of fatigue, wear, and mechanical degradation of stainless steel wire rope and general corrosion of carbon steel components of the spent fuel handling machine. Describe the inspection requirements and frequencies for the performance evaluation program to ensure that the aging effects are adequately managed.

In Section 3.9 the applicant credited the AMPs under 10 CFR Part 50 for managing the aging effects of the spent fuel handling machine.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response 3-10:

The age management program for the spent fuel handling machine was addressed in Reference 3 under 10 CFR Part 50. The aging management review of the spent fuel handling machine was discussed in Reference 3, Appendix A, Section 3.2.2. This included a discussion of the environment and a discussion of activities performed under Calvert Cliffs Performance Evaluation Program that carry out inspections of the spent fuel handling machine. These activities were assessed as providing reasonable assurance the spent fuel handling machine would be capable of performing its intended function. The NRC accepted Calvert Cliffs aging management program in Reference 4, Section 3.11.

While the current operating licenses of the Calvert Cliffs Units will expire (2034 and 2036) before the end of the requested ISFSI license renewal period (2052), even if Calvert Cliffs chooses not to further extend the operating licenses, Calvert Cliffs will remain responsible for maintaining the spent fuel pool and associated support equipment until such time that all irradiated fuel assemblies are transferred offsite. The requirements of 10 CFR 50.54(bb) will ensure components covered under 10 CFR Part 50 which support the site's ISFSI operation will continue to be appropriately maintained for the remaining duration of the ISFSI renewal period. Therefore Calvert Cliffs will commit to maintain the existing 10 CFR Part 50 aging management program for the spent fuel handling machine for the duration of the proposed ISFSI License Renewal period.

NRC RAI 3-11:

Provide the environmental conditions to which the stainless steel wire rope is exposed and rationale that the stainless steel wire rope is not subject to the aging effects of stress corrosion cracking and loss of material due to corrosion.

In Section 3.9 the applicant identified the aging effects of fatigue, wear, and mechanical degradation of stainless steel wire rope and general corrosion of carbon steel components of the spent fuel handling machine in reference to Table 16-2, Items 12 and 13 of the nuclear power plant UFSAR.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response 3-11:

The age management program for the spent fuel handling machine was addressed as part of Reference 3 under 10 CFR Part 50. The aging management review of the spent fuel handling machine was discussed in Reference 3, Appendix A, Section 3.2.2. This included a discussion of the environment and a discussion of activities performed under Calvert Cliffs Performance Evaluation Program that carry out inspections of the spent fuel handling machine. These activities were assessed as providing reasonable assurance the spent fuel handling machine would be capable of performing its intended function. The NRC accepted Calvert Cliffs aging management program in Reference 4, Section 3.11.

It is noted that while Reference 3 does not explicitly cite general corrosion as an aging mechanism for the stainless steel wire rope, the stainless steel wire rope is inspected in the credited performance evaluations as are the carbon steel components (which cited corrosion as an aging mechanism). The wire rope inspections are done every 90 days and prior to spent fuel handling machine use during the refueling outage and involve visual inspection of the rope while running the hoist through the full length of travel. In addition, as discussed in RAI A-5 response, Calvert Cliffs maintains strict chemistry control of the water in the spent fuel pool to ensure chlorides levels do not exceed acceptable limits. These actions provide reasonable assurance that the effects of aging will be adequately managed for the stainless steel wire rope.

While the current operating licenses of the Calvert Cliffs Units will expire (2034 and 2036) before the end of the requested ISFSI license renewal period (2052), even if Calvert Cliffs chooses not to further extend the operating licenses, Calvert Cliffs will remain responsible for maintaining the spent fuel pool and associated support equipment until such time that all irradiated fuel assemblies are transferred offsite. The requirements of 10 CFR 50.54(bb) will ensure components covered under 10 CFR Part 50 which support the site's ISFSI operation will continue to be appropriately maintained for the remaining duration of the ISFSI renewal period. Therefore, Calvert Cliffs commits to maintain the existing 10 CFR Part 50 aging management program for the spent fuel handling machine for the duration of the proposed ISFSI license renewal period.

Appendix A: Aging Management Programs

NRC RAI A-1:

Define significant degradation.

The second paragraph in Summary of Section A2.1 states "...has not indicated any significant degradation to any... ." The second paragraph in Section A2.3 states "...to ensure that no significant degradation to the... ." This terminology is used throughout Appendix A.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response A-1:

It is recognized that SSCs will experience some level of degradation throughout their service life. The use of the term "significant" is to distinguish between low levels of degradation (e.g., surface oxidation, micro-fissuring of concrete, paint discoloration, etc.) and the more serious degradation that could lead to loss of function.

Thus "significant degradation" refers to any aging effect of an "In Scope' SSC that would be of sufficient extent so as to impair the ability of that SSC to perform its intended safety function.

NRC RAI A-2:

Describe the inspection/surveillance requirements and frequencies for the AMPs. Include examples of reports providing the current condition and performance of ISFSI's (e.g., comparing the results from the baseline conditions and performance) inspections of structures, systems, and components (SSCs). Appendix A, Aging Management Program, discusses surveillance, monitoring, trending, and the condition and performance of ISFSI components, but information in more detail is required.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response A-2:

Table 1 below, provides a table of the aging management programs credited in Reference 1, Appendix A. This table lists the repetitive tasks and their frequencies that are used by Calvert Cliffs to monitor aging mechanisms during the license extension period.

Table 1

Program	Rep Task ID #	Frequency	Inspection
Site-Specific ISFSI Aging Management	RT 01012013	Annually	• To perform ISFSI HSM Rebar Inspection and look for spalled and cracking concrete per checklist
	Mounting Video around the ISFSI Site	Constant surveillance	• To check the HSM inlet and outlet screens not to be obstructed
Transfer Cask Aging Management	RT 01012007	Annually	• To perform PT examination on ISFSI transfer cask trunnions, and all attachment welds to trunnions and to follow up with UT if indications are found • To examine entire trunnion surface including approx. 2" on cask surface
Transfer Cask Lifting Yoke Aging Management	RT 01012009	Annually	• To inspect the lifting beams and hooks are straight and there is no distortion from previous loading • To perform visual inspection on the area around the lifting pin hole, at the shoulders where the lifting beams connect to the lifting hooks, area beneath and around the palms of the hooks • To perform MT exams on the following areas: inside/outside of the lifting pin hole, inside of the palm of the lifting hooks
Cask Support Platform Aging Management	Chemistry Sampling Program	Monthly	• Conduct chloride sampling of spent fuel pool water

Calvert Cliffs provided a historical review of corrective actions taken against applicable components in our response to the NRCs request for supplemental information #3 (Reference 5). Listed below is a summary of the results from the latest conduct of the repetitive tasks listed in Table 1 above.

ISFSI AMP: Repetitive Task 01012013, Last conducted: 04/21/2011

This repetitive task performs a visual inspection of the HSMs concrete structures looking for exposed rebar. This inspection is performed annually using a site approved checklist. This inspection identified areas of minor concrete cracking, spalling, and mineral deposits on the foundation slab. However there was no sign of water intrusion or of exposed rebar. Inspection of the ground floor slab did not reveal any environmental degradation, corrosion of rebar or water intrusion.

Transfer Cask AMP: Repetitive Task 01012007, Last conducted: 09/22/2010

This repetitive task performs penetrant testing on the four Transfer Cask trunnions. The repetitive task is performed annually. It includes examination of the entire trunnion surface including 2 inches on the cask surface. In addition to the penetrant testing performed by this repetitive task, a visual inspection of the external transfer cask and cask lid surfaces is performed as part of a site procedure prior to using the transfer cask to move a DSC. This visual inspection looks for signs of corrosion in order to provide confidence the Transfer Cask remains able to perform its intended function. During this inspection no areas of degradation that would impact the ability of the Transfer Cask to perform its intended function were observed.

Transfer Cask Lifting Yoke AMP: Repetitive Task 01012009, Last conducted: 9/15/2010

This repetitive task is a detailed inspection of the Transfer Cask Lifting Yoke. The inspection consists of both magnetic particle testing and visual testing of the yoke. The repetitive task is performed annually. As part of the inspection, yoke dimensions are checked and areas of flaking paint are identified and repainted. During this latest inspection no issues that would impact the ability of the Transfer Cask Lifting Yoke to perform its intended function were identified.

Cask Support Platform AMP: Chemistry Sampling Program

In order to prevent loss of material and cracking for the stainless steel cask support platform that sits in the spent fuel pool, Calvert Cliffs relies upon maintaining strict chemistry control of the spent fuel pool water. On a monthly basis Calvert Cliffs samples the spent fuel pool for chlorides. Calvert Cliffs has established a strict threshold for chloride levels of < 100 parts per billion. This threshold is well below the EPRI PWR Primary Guidelines value which is set at <150 parts per billion. A review of the last three years of sample results showed no occurrence where the site threshold for chlorides was exceeded.

<u>NRC RAI A-3:</u>

Provide the review results of the CAP mentioned in the operating experience program element of Section A2.1. HSM AMP, which indicates that any deficiencies identified for the HSM have been administrative and were not related to the effects of aging. Include the records associated with the instance where minor cracking was noted on top of the HSMs that required cosmetic crack repair. Provide an evaluation regarding the root cause of the concrete cracking, and justify why this condition will not lead to accelerated component degradation during the license renewal period.

In Section A2.1 the applicant stated that it has reviewed its CAP and found that any deficiencies identified for the HSM have been administrative and were not related to the effects of aging. However, the applicant did not provide detailed discussions of its plant-specific operating experience.

This is required to evaluate compliance with 10 CFR 72.120 and 10 CFR 72.172.

CCNPP Response A-3:

In Reference 5 Calvert Cliffs provided, in its response to request for supplemental information 3, a list of condition reports written in regards to HSM. These issues were mostly identified during performance of the periodic repetitive task that performs a visual inspection of the HSMs concrete surfaces. Each of these identified issues was entered into Calvert Cliffs corrective action process and maintenance work orders were developed as applicable. Maintenance work orders are then prioritized and are worked based on their assigned priority and as manpower resources are available. The issues listed in Reference 5 were evaluated as not impacting the HSMs ability to perform its safety function. Based on this prioritization most of these work orders have not been performed. However conduct of the periodic visual inspection repetitive task helps provides ongoing monitoring of the identified conditions. To provide a more thorough assessment of the current conditions, Calvert Cliffs commits to conduct an engineering evaluation of the identified concrete degradations, performed by a qualified structural engineer.

NRC RAI A-4:

Provide copies of applicable guidance and direction for maintaining a suitable environment that prevents the occurrence of loss of material due to corrosion for wetted surfaces, as described in the Preventive Actions elements of the AMPs for transfer cask, transfer cask lifting yoke, and cask support platform. Define the meaning of the term ~a suitable environment,~ and explain how the environment is maintained to prevent the aging effects.

In Sections A2.2, A2.3, and A2.4 the applicant stated that the AMPs for transfer cask, transfer cask lifting yoke, and cask support platform include guidance and direction for maintaining a suitable environment that prevents the occurrence of loss of material due to corrosion for wetted surfaces. However, the applicant did not provide detailed discussions of the preventive actions.

This is required to evaluate compliance with 10 CFR 72.120.

CCNPP Response A-4:

The chemistry of the spent fuel pool is maintained to control leachable chlorides and fluorides in order to maintain a non-corrosive environment for the stainless liner, fuel racks, and components and cask support platform and other structures stored in the pool.

The transfer cask and lifting yoke are stored in a dry inside environment while not in use for fuel moves. During fuel moves the cask and yoke are washed down with demineralized water upon removal from the pool water. These steps are contained within the site procedure covering the loading and transfer of a DSC from the spent fuel pool.

In addition to the above steps, regular inspections of the transfer cask and transfer cask lifting yoke are performed under rep tasks as shown in our response to RAI A-2 above.

NRC RAI A-5:

Provide plant-specific operating experience including chloride sampling records and associated corrective actions associated with instances of unsatisfactory degradation that were entered in the CAP.

Explain how unsatisfactory degradation of the cask support platform described in Section A2.4, Cask Support Platform AMP, is determined, given that the Cask Support Platform AMP does not include visual inspections of the cask support platform.

In Section A2.4 the applicant stated that any out-of-specification results and unsatisfactory degradation are entered in its CAP for resolution. However, the applicant did not provide detailed discussions of its plant-specific operating experience.

This is required to evaluate compliance with 10 CFR 72.120 and 10 CFR 72.172.

CCNPP Response A-5:

The Cask Support Platform Aging Management Program relies on our Chemistry Control Program to monitor chloride levels in the spent fuel pool to ensure conditions that would be conducive to the onset or propagation of loss of material and cracking due to pitting and/or stress corrosion in this stainless steel platform do not occur. Under our Chemistry Control Program, a chloride sample of the spent fuel pool water is taken monthly and compared to the target threshold value (<100 ppb). The target value selected by Calvert Cliffs is more conservative than the target recommendation specified in the EPRI PWR Primary Guidelines of <150 ppb.

January 2010 – June 2011 Spent fuel pool chloride data:

Spent fuel pool chloride ranged from less than 5.0 ppb to 13.1 ppb. 5.0 ppb corresponds to the limit of detection. Spent fuel pool chloride average for this period is 5.8 ± 1.7 ppb.

A review of the spent fuel pool chloride sample values taken during the last three years showed no instances where our conservative threshold value was exceeded.

Furthermore, the spent fuel pool ion exchanger effluent is sampled and analyzed once a month. Trending is routinely performed to identify any degrading trends. The spent fuel pool ion exchanger resin is replaced annually prior to the refueling outage to ensure adequate cleanup capacity exists.

A review of plant operating experience for the cask support platform was conducted as part of our license renewal application. This review did not identify any occurrences of unsatisfactory degradation associated with the cask support platform. These results further support that maintaining and monitoring of the spent fuel pool water conditions provides reasonable assurance that the cask support platform will be able to continue to perform its intended function throughout the license renewal period.

Appendix B: Time-Limited Aging Analysis

NRC RAI B-1:

Provide a description of the expression and assumptions used in Appendix 8, ISFSI Time-Limited Aging Analysis Report, Section 4.1 for the calculation of the estimated total scalar neutron flux and the technical basis for the selection of parameter values used in the calculation.

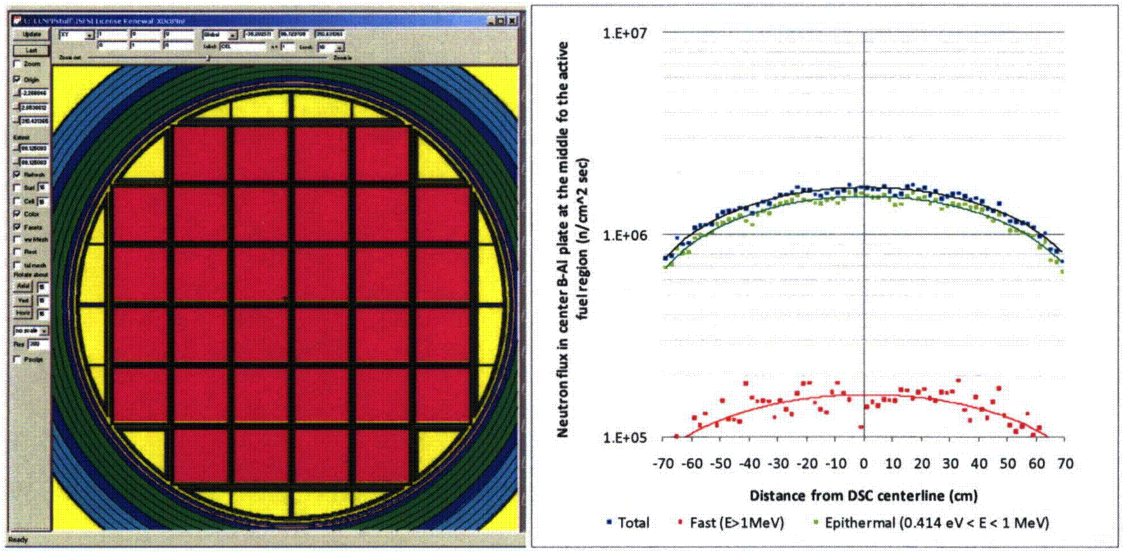
In the Appendix 8, ISFSI Time-Limited Aging Analysis Report, Section 4.1, ISFSI Materials and Analyses Review of Poison Plates for the NUHOMS-32P, an approximation is used to estimate the total scalar neutron flux based upon a calculated scalar neutron flux. The applicant is requested to provide a description of the expression and assumptions used in the calculation of the estimated total scalar flux as well as a technical basis for the selection of parameter values used in the calculation.

This is required to evaluate compliance with 10 CFR 72.24(d).

CCNPP Response B-1:

The method utilized in Reference 1, Appendix B, Section 4.1, simply divided the design basis assembly neutron source for a NUHOMS-32P DSC by the inner surface area of a NUHOMS-32P DSC fuel cell. The use of the word scalar refers to the fact that neutron direction is ignored in this estimate and all source neutrons are assumed to pass through the fuel cell wall (i.e., axial leakage is ignored). To provide a second check of the approximation used in Reference 1, Appendix B, Section 4.1, Calvert Cliffs performed an independent check for this response using an in-house MCNP5 model of a NUHOMS-32P DSC containing the design basis neutron source from Calvert Cliffs Calculation CA06721, Section 6.5 (see ADAMS Accession Number ML091680542). The MCNP5 model was based on the transfer cask model used in Calvert Cliffs Calculation CA06750 (see ADAMS Accession Number ML091680544), modified to include an explicit model of the basket with homogenized fuel. A neutron flux mesh tally over the center poison plate was utilized to calculate the flux over the entire length of the poison plate. The results at the centerline of the active fuel region suggest that the method utilized in Reference 1, Appendix B may not reflect the peak neutron flux, which is calculated to be 1.69E6 n/cm^2s at the center of the plate using MCNP5 model (as shown in Figure 1 below). The energy distribution of the flux is 93% epithermal (0.414 eV < E < 1 MeV) and 7% fast (E > 1 MeV), with the thermal flux being negligible. This increases the 60 year fluence to 3.20E15 n/cm^2 using the same assumptions as in Reference 1, Appendix B, Section 4.1 (no decay in design basis source over 60 years). Note, however, that this does not alter the Section 4.1 conclusion that the 60 year fluence has a negligible impact on boron content of the plates, since 99.9993% of the boron would remain after 60 years even if every neutron was assumed to be absorbed by B-10. If actual B-10 reaction rates were considered based on the energy distribution of the flux, the depletion would be expected to be several orders of magnitude less.

Figure 1



NRC RAI B-2:

Provide the technical details and basis required to clarify the differences in the finite element analyses that produced the outputs labeled Cask Body and Cask Body Model provided in documents TransNuclear (TN) Calculation 1095·6 and TN Calculation 1095-16.

In Appendix B, ISFSI Time-Limited Aging Analysis Report, Section 4.3, Transfer Cask Fatigue Evaluation, the applicant refers to the transfer cask fatigue evaluation documented in AREVA Calculation 10955-0203. Within AREVA Calculation 10955-0203, the applicant uses the maximum *temperatures taken from finite element analyses provided in documents TN Calculation 1095-16 and TN Calculation 1095-6.*

In the document TN Calculation 1095-16, the applicant obtains the maximum temperature (370.792 °F) from the finite element analysis output labeled Cask Body Model. In the document TN Calculation 1095-6, the applicant obtains the maximum temperature (355.321 °F) from the finite element analysis output labeled Cask Body; however, in the same document, another finite element analysis output labeled Cask Body Model shows a maximum temperature of 459.932 °F. In AREVA Calculation 10955-0203, using the fifth criterion (e) of ASME NC-3219.2, the maximum temperature should not exceed the calculated value of 435 °F. Therefore, if correct temperature from the finite element analysis should be 459.932 °F, this would violate the fifth criterion (e) of ASME NC-3219.2. The applicant is requested to clarify the difference between the finite element analyses that produced the outputs labeled Cask Body and Cask Body Model by providing the technical details and basis for the selection of each of the finite element analysis outputs used for obtaining the maximum temperatures used in the transfer cask fatigue calculations.

This is required to evaluate compliance with 10 CFR 72.24(d).

CCNPP Response B-2:

TransNuclear Calculation 1095-6 determines component temperatures of the NUHOMS-32P system during transfer with an ambient temperature of 103°F and an insolation of 127 Btu/hr-ft^2. TransNuclear Calculation 1095-16 determines component temperatures of the NUHOMS-32P system during transfer with an ambient temperature of -3°F and no insolation.

Both calculations employ two finite element models: cask body model and basket model. The cask body model consists of the cask body, lid, ram plate, and canister. The basket model consists of the canister, basket, and fuel assemblies.

TransNuclear Calculation 1095-6 provides a temperature distribution plot for the entire "cask body model" (cask body, lid, ram plate, canister) showing a maximum temperature 459.932°F and provides the following component temperature plots: canister (showing a maximum temperature of 459.932°F), cask body (showing a with maximum temperature of 355.321°F) and cask lid (showing a maximum temperature of 355.321°F).

TransNuclear Calculation 1095-16 provides plots of temperature distributions for only the entire "cask body model" showing a maximum temperature of 370.792°F. Temperature distribution plots for the individual cask components are not provided in that calculation.

Both calculations provide detailed lists of the maximum temperatures for each component of "cask body model" in Section 6.0 that can be referred to in case the maximum component temperature is needed. In particular, TransNuclear Calculation 1095-6 determines maximum temperature for cask body as 355°F and maximum canister temperature as 460°F, while TransNuclear Calculation 1095-16 determines maximum "cask body" temperature as 233°F, and canister temperature as 371°F.

The above information shows that the maximum temperature of 459.932°F on the temperature distribution plot of the "cask body model" in TransNuclear Calculation 1095-6 is the maximum temperature of the canister component, not the transfer cask.

AREVA Calculation 10955-0203 used as a supporting reference material two plots: Figure 1 - plot of temperature profile for "cask body" (taken from TransNuclear Calculation 1095-6) and Figure 2 - plot of temperature profile for the "cask body model" (taken from TransNuclear Calculation 1095-16).

These two plots were referenced to and utilized in AREVA Calculation 10955-0203 to estimate the envelope for the maximum temperature difference between any two adjacent points of the transfer cask to prove that criterion (c) and criterion (d) of ASME NC-3219.2 are satisfied. The envelope for the maximum temperature difference between any two adjacent points of the transfer cask was derived from the contour plot temperature increment size of these plots. TransNuclear Calculation 1095-16 provided the temperature distribution for the entire "cask body model." Since the plot included transfer cask and canister components temperature distribution it was deemed sufficient and adequate to determine the envelope for the maximum temperature difference between any two adjacent points of the transfer cask. TransNuclear Calculation 1095-6 explicitly provided the plot of temperature distribution of the "cask body" component. Thus the temperature differences for the transfer cask were determined using the above mentioned figures.

In the case of the examination of criterion (e) of ASME NC-3219.2, the critical area, for which the temperature limit of 435°F is to be applied, was determined to be the interface between the trunnion sleeve (material SA-182 Type F304N) and the trunnion (material SA-564 Type 630PH). In such a case, the maximum temperature for "cask body" component can serve as an adequate conservative estimate of maximum temperature range for the critical area. For the 103°F ambient temperature case, the maximum temperature for the cask body is 355°F (TransNuclear calculation 1095-6, section 6.0), while for the -3°F ambient temperature case the maximum temperature for the cask body is 233°F (TransNuclear Calculation 1095-16, section 6.0). These conservative estimates for maximum temperature range of cask body component are significantly below the calculated limit of 435°F. Therefore AREVA Calculation 10955-0203 concluded, referring to TransNuclear Calculations 1095-6 and 1095-16 in general – not to Figure 1 or Figure 2, that maximum temperature range for transfer cask is significantly below the applicable limit.

Other

NRC RAI O-1:

The following editorial mistake in specification of units for maximum exposure in Section 3.2.1, Description of Irradiated Fuel Assemblies Subcomponents, Fuel Rods (Cladding, End Caps/Plugs) should be corrected:

"A small number of assemblies in storage have up to five solid stainless steel replacement rods, with a maximum exposure ≤ 40,000 MDd/MTU, in place of fuel rods."

This is an editorial correction.

CCNPP Response O-1:

Yes, this was a typographical error. The maximum exposure stated in this sentence of Section 3.2.1 should be stated as "≤ 40,000 MWd/MTU" vice "≤ 40,000 MDd/MTU".

NRC RAI O-2:

Provide an evaluation demonstrating the adequacy of the SSCs identified in the Scoping Evaluation in assuring ready retrieval of spent fuel for further processing or disposal for the duration of the licensing *period and a description of how long-term effects that could affect the ready retrieval of spent fuel for the duration of the licensing period are addressed for each relevant SSC.*

Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal for the duration of the licensing period, according to 10 CFR 72.122(l) and 10 CFR 72.236(m). Although the LRA makes reference to retrievability in several places (e.g. "The fuel cladding provides a confinement barrier, and its structural integrity is necessary to maintain a favorable geometry and for retrieval", "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 staff was not able to identify a discussion explicitly specifying long-term effects that may affect the ready retrieval of spent fuel for further processing or disposal for the duration of the licensing period. The licensee should provide an evaluation demonstrating the adequacy of the SSCs identified in the Scoping Evaluation in assuring ready retrieval of spent fuel for further processing or disposal for the duration of the licensing period. The licensee should also provide a description of how long-term effects that could affect SSCs identified as inscope in the LRA Scoping Evaluation and relied upon for retrieval of spent fuel are addressed to prevent any potential retrievability issue for continued operation during the license renewal period.

This is required to evaluate compliance with 10 CFR 72.122(1) and 10 CFR 72.236(m)

CCNPP Response O-2:

The capacity of the existing hydraulic ram system is 80 kips (k) and the maximum weight of the dry shielded cask is 91 k. The force required to retrieve a cask whose rails have been treated with a graphite based dry film lubricant is 4.5 k. Using a conservative static friction coefficient for dry steel on steel of 0.8, the maximum force needed to remove the heaviest cask without the benefit of any lubricant is 91 k x 0.8 = 72.8 k which is less than the capacity of the hydraulic ram system of 80 k. Therefore, retrievability of spent fuel from the HSMs is assured.

NRC RAI O-3:

Provide confinement/dose analyses for the casks located at the ISFSI.

The Technical Specifications indicate that the top shield plug closure and the siphon and vent port cover welds are leak tested to 10-4 atm-cc/sec (presumably 1E10-4 atm-cc/sec). Section 3.3.2.1 of the Updated Safety Analysis Report (USAR) states that bottom, girth, and longitudinal welds were leak tested with soap bubble film, which has a nominal test sensitivity of 10-3 ref-cm3/sec (American National Standards Institute) (ANSI)-N14.5). Since these leak rates are greater than leaktight criterion (1E10·7 ref-cm3/sec, per ANS-N14.5), a confinement analysis and the resulting doses should be provided for normal, off-normal, and accident conditions. Total dose as a result of the canisters' leakage rate is a function of number of casks, canister leak rate, percentage of rod failures, fraction of gases, volatiles, fines, and crud released. Guidance for such a calculation is presented in NUREG-1536, Rev. 1 (Section 5). [Note: The USAR presented a limited confinement analysis for an accident condition (Section 8.2.8 of the USAR), but only assumed Kr-85 gas release from a single canister.]

This information is required to evaluate compliance with 10 CFR 72.104 and 10 CFR 72.106.

CCNPP Response O-3:

Comparison of tested leak rates versus the Technical Specification 3.2.2.2 requirements is included in our response to RAI O-4 below.

The licensing basis confinement analysis for the Calvert Cliffs ISFSI is summarized in Calvert Cliffs ISFSI USAR Section 8.2.8 for the NUHOMS-24P canister and Section 12.8.2.8 for the NUHOMS-32P canister (see ADAMS Accession Numbers ML102590455 and ML102590476, respectively), and involves a non-mechanistic instantaneous (puff) release to the environment of the gap inventory of Kr-85 fission gas from all fuel rods contained in a single canister (1.39E3 Ci for the NUHOMS-24P DSC and 9.13E4 Ci for the NUHOMS-32P DSC). The fraction of Kr-85 released from the pellet to the gap is calculated using the American Nuclear Society 5.4-82 method to be 2.1% for the NUHOMS-24P DSC (47 GWd/MTU maximum burnup) and 9.35% for the NUHOMS-32P DSC (52 GWd/MTU maximum burnup). An atmospheric dispersion factor (X/Q) of 3.0E-4 sec/m^3, was used in calculating the maximum potential doses at the 3900 foot (1189 m) distance to the controlled area boundary. The X/Q was calculated using Regulatory Guide 1.145 methodology assuming G stability and 1 m/s wind speed. The resulting calculated doses for the NUHOMS-24P DSC are 0.1 mrem and 17.8 mrem for the maximum off-site total body and skin doses, respectively. The calculated total body and skin doses for the NUHOMS-32P DSC are 0.65 and 109.6 mrem, respectively.

In response to RAI O-4 below, Calvert Cliffs has also performed a calculation using the expanded inventory and release fraction requirements of Reference 4. Source terms for a design basis 660 watt assembly were obtained from Calvert Cliffs Calculation CA06721 Section 6.5 (see ADAMS Accession Number ML091680542), which was approved by the NRC under Calvert Cliffs ISFSI License Amendment 9. The crud Co-60 activity at discharge was based on the 140 $\mu Ci/cm^2$ recommended for PWRs in Reference 4, Table 5-2, a total fuel rod surface area of 2.31E5 cm^2 for a CE 14x14 assembly, and was decay corrected to the time indicated for the design basis source in Calvert Cliffs Calculation CA06721 (16y for neutron, 7y for gamma). The dose conversion factors (DCFs) from Federal Guidance Reports (FGR) 11 (Inhalation) and 12 (Submersion) were utilized in these calculations. The FGR 11 DCFs used for each isotope were the bounding lung clearance class for each organ, with the exception of uranium isotopes for which the "Year" lung clearance class was used based on the fact that FGR 11, Table 3 indicates this is the appropriate class to use for UO_2. Fuel gap-to-DSC release fractions for all cases were those recommended in Reference 4, Table 5-2, which were taken directly from NUREG/CR-6487.

The above inputs and methods were utilized to perform the normal, off-normal, and accident confinement analyses required by Reference 4. The details and results from each are discussed below:

Normal

As required by Reference 4, the confinement analysis for normal conditions is based on normal release from all DSCs stored in the ISFSI at the Calvert Cliffs ISFSI Technical Specification 3.2.2.2 leak rate of 1E-4 atm-cc/sec (0.07% release from 171 ft^3 DSC internal void space in 8760 hours). Each DSC is assumed to contain 1% of fuel failed during storage as required by Reference 4, Table 5-2. Based on the response to RAI O-5, only the first three phases of the Calvert Cliffs ISFSI are considered (48 NUHOMS-24P DSCs, and 24 NUHOMS-32P DSCs) which store a total of 1920 fuel assemblies, since only those phases would be subject to our current Technical Specification 3.2.2.2 leak test criteria.

All of the gaseous radionuclides (H-3, C-14, Cl-36, Kr-81, Kr-85, I-129, Xe-127) released from the fuel gap were considered available for release from the DSC. Based on NRC Regulatory Guide 1.183, 5% of the cesium was considered to be in the elemental or organic chemical form, and available for release from the DSC, and the remaining 95% was in the form of cesium iodide particulate. Release of particulate (cesium, fuel fines, crud) from the DSC to the environment was not considered credible for this scenario since the release rate from the canister is so low under this scenario that sufficient time exists for gravitational settling to remove most particulate from the internal DSC atmosphere. Furthermore, based on the distribution of irradiated UO_2 particle sizes given in NUREG-1320 (Figures 4.12-4.14; 99.9% > 10μm) these materials would also be too large to escape though the size penetrations that could lead to a 1E-4 cc/sec release rate. Crud particle size is generally smaller than fuel fines (0.2-20 μm based on

SAND88-1358, Figure 5) however, detailed examination of a CASTOR V/21 cask in storage at Idaho National Laboratory found no evidence of crud in the gas samples taken from the cask, and no evidence of major crud spallation from the fuel rod surfaces after 14 years of storage (see EPRI TR-1002882 or INEEL/EXT-01-00183). No credit was taken for retention of material released from the DSC within the HSM. Table 1 below summarizes the release fractions applied for normal releases.

Table 1 – Release Fractions for Normal Conditions

Release Fractions	Fuel to Cask	Cask to HSM	HSM to Env	Total
Gases	0.3	1	1	0.3
Volatiles	2.0E-04	0.05	1	1.0E-05
Fines	3.0E-05	0	1	0
Crud	0.15	0	1	0

Table 2 below summarizes the doses calculated for the normal release scenario. Doses assume constant occupancy at the site boundary for 8760 hours. Using Regulatory Guide 1.145, D-stability, and 5 m/s wind speed, as recommended by Reference 4, an X/Q of 2.0E-5 sec/m^3 is determined for the 3900 foot distance to the controlled area boundary. Note that the above release assumptions are entirely hypothetical and actual air monitoring performed as part of the Radiological Environment Monitoring Program has detected no airborne releases to date from the ISFSI.

Table 2 – Normal Confinement Annual Doses

CA06721 Source	Dose Component	GONADS	BREAST	LUNGS	RED MARROW	BONE SURFACE	THYROID	REMAINDER	EFFECTIVE	SKIN
Neutron	DDE (mrem)	6.89E-05	7.88E-05	6.75E-05	6.46E-05	1.27E-04	6.96E-05	6.44E-05	7.01E-05	6.46E-03
	CEDE (mrem)	1.22E-03	1.12E-03	1.38E-03	1.17E-03	1.13E-03	3.32E-03	1.26E-03	1.29E-03	0.00E+00
	Total (mrem)	1.29E-03	1.20E-03	1.45E-03	1.24E-03	1.26E-03	3.39E-03	1.33E-03	1.36E-03	6.46E-03
Gamma	DDE (mrem)	1.08E-04	1.24E-04	1.06E-04	1.01E-04	2.00E-04	1.09E-04	1.01E-04	1.10E-04	1.05E-02
	CEDE (mrem)	1.42E-03	1.29E-03	1.50E-03	1.35E-03	1.30E-03	2.92E-03	1.47E-03	1.46E-03	0.00E+00
	Total (mrem)	1.53E-03	1.41E-03	1.60E-03	1.45E-03	1.50E-03	3.03E-03	1.57E-03	1.57E-03	1.05E-02
Max Total (mrem)		**1.53E-03**	**1.41E-03**	**1.60E-03**	**1.45E-03**	**1.50E-03**	**3.39E-03**	**1.57E-03**	**1.57E-03**	**1.05E-02**

Based on Calvert Cliffs Calculation CA06751 (see ADAMS Accession Number ML091680545) Table 6-9, the annual direct dose at this location from a fully loaded 120 HSM ISFSI would be less than 4.6E-2 mrem (5.3E-6 mrem/hr x 8760 hours). Thus, the normal annual ISFSI doses from hypothetical releases and direct radiation are a small fraction of the 10 CFR 72.104(a) requirement that the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ.

Off-Normal

As required by Reference 4, the confinement analysis for off-normal conditions is based on normal release from a single DSC stored in the ISFSI at Technical Specification 3.2.2.2 leak rate of 1E-4 atm-cc/sec. The DSC is assumed to contain 10% of fuel failed during storage as required by Reference 4. The DSC leak rate used in the calculation was increased by a factor of 10 to conservatively bound the impact of increased DSC internal pressure due to the failed fuel rods assumed (0.65% release from 171 ft^3 DSC internal void space in 8760 hours). The exposure duration, release fractions and atmospheric dispersion factors (X/Q) are the same as those discussed above for normal conditions. Table 3 below summarizes the doses calculated for the off normal release scenario. As noted previously the above release assumptions are entirely hypothetical and actual air monitoring performed as part of the Radiological Environment Monitoring Program has detected no airborne releases to date from the ISFSI.

Table 3 – Off-Normal Confinement Annual Doses

CA06721 Source	Dose Component	GONADS	BREAST	LUNGS	RED MARROW	BONE SURFACE	THYROID	REMAINDER	EFFECTIVE	SKIN
Neutron	DDE (mrem)	1.14E-04	1.31E-04	1.12E-04	1.07E-04	2.11E-04	1.16E-04	1.07E-04	1.17E-04	1.07E-02
	CEDE (mrem)	2.04E-03	1.87E-03	2.30E-03	1.95E-03	1.89E-03	5.51E-03	2.10E-03	2.15E-03	0.00E+00
	Total (mrem)	2.15E-03	2.00E-03	2.41E-03	2.06E-03	2.10E-03	5.63E-03	2.21E-03	2.27E-03	1.07E-02
Gamma	DDE (mrem)	1.80E-04	2.06E-04	1.76E-04	1.68E-04	3.32E-04	1.82E-04	1.68E-04	1.83E-04	1.74E-02
	CEDE (mrem)	2.36E-03	2.14E-03	2.49E-03	2.25E-03	2.16E-03	4.86E-03	2.44E-03	2.42E-03	0.00E+00
	Total (mrem)	2.54E-03	2.35E-03	2.67E-03	2.41E-03	2.49E-03	5.04E-03	2.61E-03	2.60E-03	1.74E-02
Max Total (mrem)		**2.54E-03**	**2.35E-03**	**2.67E-03**	**2.41E-03**	**2.49E-03**	**5.63E-03**	**2.61E-03**	**2.60E-03**	**1.74E-02**

Based on Calvert Cliffs Calculation CA06751 (see ADAMS Accession Number ML091680545) Table 6-9, the annual direct dose at this location from a fully loaded 120 HSM ISFSI would be less than 4.6E-2 mrem (5.3E-6 mrem/hr x 8760 hrs). Thus, the off-normal annual ISFSI doses from hypothetical releases and direct radiation are a small fraction of the 10 CFR 72.104(a) requirement that the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ.

Accident

For consistency with the current design basis DSC leakage accident analysis, this updated accident confinement analysis assumes a release from a single NUHOMS-32P DSC (bounds a NUHOMS-24P DSC). The DSC is assumed to contain 100% of fuel failed during storage as required by Reference 4. While the internal DSC pressure which would result from such a failure is below the 100 psig design pressure of the NUHOMS-32P DSC (and thus failure from this condition alone would not be expected) for the purpose of this analysis a non-mechanistic failure in the form of a 1 mm^2 hole is assumed. This size opening is consistent with a pit or crack type penetration which might be associated with the unmitigated effects of aging, and would lead to depressurization of the DSC in approximately 11 hours (from accident pressure to atmospheric). As with the design basis analysis, this can be treated as a puff for the purposes of the 30-day occupancy assumption required by Reference 4.

All of the gaseous radionuclides (H-3, C-14, Cl-36, Kr-81, Kr-85, I-129, Xe-127) released from the fuel gap were considered available for release from the DSC. DSC-to-HSM release fractions were taken from NUREG/CR-6672, where cask-to-environment release fractions of 0.0008 for CsI (volatiles) and 0.02 for other particulates is given in Table 7.19 for a 60 mph impact of a mechanically sealed rail cask resulting in a 1 mm^2 leak area. Based on the distribution of irradiated UO_2 particle sizes given in NUREG-1320 (Figures 4.12-4.14) only 10% of the population of fuel particles that would fit through a 1 mm^2 opening were less than or equal to 10 μm size normally considered the upper bound of the respirable range and also likely to stay airborne for more than a few hours (i.e., the time required for the DSC to blow down). Therefore, an additional factor of 0.1 was applied only to fuel fines to address this factor. This treatment of particulate releases is considered conservative as the product of the above release fractions (see Table 4 below) is greater than that cited by the NRC staff in the original March 1991 NRC Environmental Assessment for the Calvert Cliffs ISFSI (see ADAMS Accession Number ML022550053), where a total release fraction of 5E-10 for particulates is used in Tables 6.3 and 6.4. No credit was taken for retention of material released from the DSC within the HSM. Table 4 below summarizes the release fractions applied for accident releases.

Table 4 – Release Fractions for Accident Conditions

Release Fractions	Fuel to Cask	Respirable	Cask to HSM	HSM to Env	Total
Gases	0.3	1	1	1	0.3
Volatiles	2.00E-04	1	0.0008	1	1.60E-07
Fines	3.00E-05	0.1	0.02	1	6.00E-08
Crud	1	1	0.02	1	0.02

Table 5 below summarizes the doses calculated for the accident release scenario. Doses assume constant occupancy at the site boundary for 720 hours. Using Regulatory Guide 1.145, F-stability, and 1 m/s wind speed, as recommended by Reference 4, an X/Q of 1.8E-4 sec/m^3 is determined for the 3900 foot distance to the controlled area boundary.

Table 5 – Accident Confinement 30-Day Doses

CA06721 Source	Dose Component	GONADS	BREAST	LUNGS	RED MARROW	BONE SURFACE	THYROID	REMAINDER	EFFECTIVE	SKIN
Neutron	DDE (rem)	0.001	0.002	0.001	0.001	0.003	0.001	0.001	0.001	0.149
	CEDE (rem)	0.067	0.007	0.537	0.358	4.177	0.058	0.161	0.262	0.000
	Total (rem)	0.069	0.009	0.538	0.359	4.180	0.059	0.163	0.264	0.149
Gamma	DDE (rem)	0.002	0.003	0.002	0.002	0.004	0.002	0.002	0.002	0.241
	CEDE (rem)	0.044	0.010	0.487	0.225	2.456	0.047	0.099	0.174	0.000
	Total (rem)	0.046	0.013	0.489	0.227	2.460	0.049	0.101	0.176	0.241
Max Total (rem)		**0.069**	**0.013**	**0.538**	**0.359**	**4.180**	**0.059**	**0.163**	**0.264**	**0.241**

Based on Calvert Cliffs Calculation CA06751 (see ADAMS Accession Number ML091680545) Table 6-9, the annual direct dose at this location from a fully loaded 120 HSM ISFSI would be less than 3.8E-3 mrem (5.3E-6 mrem/hr x 720 hours). Thus, the 30-day ISFSI accident doses are below the 10 CFR 72.106(b) requirement that any individual located on or beyond the nearest boundary of the controlled area not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem, or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem, 15 rem dose equivalent to the lens of the eye (determined by the sum of the skin and TEDE dose), or 50 rem shallow dose equivalent to the skin or any extremity.

NRC RAI 0-4:

Provide confirmation that canisters loaded after November 2012 will satisfy confinement effectiveness during the lifetime of the license.

Canisters are leak tested per ANSI N14.5 to determine confinement effectiveness (NUREG-1536). Fabrication leak rate tests of the entire confinement boundary, including welds and base material (except the lid closure weld if per Interim Staff Guidance-18), should be performed. The leak rate of the canisters, coupled with the release fractions associated with gases, volatiles, fines, and crud, provide a key component of dose at the site boundary.

This information is required to evaluate compliance with 10 CFR 72.236, 10 CFR 72.104, and 10 CFR 72.106.

CCNPP Response 0-4:

While the current Calvert Cliffs ISFSI Technical Specification 3.2.2.2 requires that the helium leak rate for the top shield plug closure weld, and the siphon and vent port cover welds, not exceed

1E-4 atm-cc/sec, internal administrative requirements have in many cases been more stringent, especially for more recently loaded canisters.

A review of the procurement specifications for the later NUHOMS-24P DSCs indicates that DSC shell confinement boundary welds were required to be shop tested to demonstrate a maximum helium leak rate of 1E-5 atm-cc/sec under an internal helium pressure of at least 10 psig. However for the NUHOMS-24P DSC closure welds, the acceptance criteria that was applied was simply the ISFSI Technical Specification 3.2.2.2 requirement, although a review of helium leak test results for a small population of the NUHOMS-24P DSCs, found that all reported closure weld leak rates were also below 1E-5 atm-cc/sec.

A review of the procurement specifications for all the NUHOMS-32P canisters used to date (installation of NUHOMS-32P DSCs began in November 2005) indicates that DSC shell confinement boundary welds were required to be shop tested to demonstrate a maximum helium leak rate of 1E-7 atm-cc/sec under an internal helium pressure of at least 22.5 psig (a temporary mechanical seal was used for the top plug closure). Concurrent with the installation of NUHOMS-32P DSCs, Calvert Cliffs began using improved helium leak detection equipment that is capable of detecting helium leak rates below 1E-7 atm-cc/sec for NUHOMS-32P DSC closure operations. A review of leak test results indicates that the highest reported helium leak rate to date for a NUHOMS-32P DSC during closure (top shield plug and the vent and siphon port cover welds) is less than 3E-8 atm-cc/sec.

Calvert Cliffs will commit to changing the ISFSI Technical Specification 3.2.2.2 helium leak rate requirement for the top shield plug closure weld, and the siphon and vent port cover welds, from 1E-4 atm-cc/sec to 1E-7 atm-cc/sec following the loading of all existing modules. While it is anticipated all existing modules will be used prior to November 2012, Calvert Cliffs would prefer to tie the change to after the completion of loading of all existing ISFSI modules and their associated DSCs rather than to a specific date.

Response to Request for Supplemental Information

NRC RAI RSI-1:

Justify the radiation survey measurements recorded in 2001 from the CCNPP ISFSI Area Radiation Surveys.

In the Response to Request for Supplemental Information dated February 10, 2011, the applicant provided copies of representative dose rate surveys in Enclosure 5, ISFSI Area Radiation Surveys, The survey documented for 2001 contains dose rate measurements for the HSM bottom vents that appear to be higher than for previous or subsequent years. Further, on the survey document, the surveyors indicate numbers that appear to be high for the bottom vent. Please provide a justification for the observed measurements for 2001.

This is required to evaluate compliance with 10 CFR 72.126.

CCNPP Response RSI-1:

A review of the radiation survey measurements was conducted and as indicated the contact readings from the 2001 survey were elevated in regards to the levels in both the years that preceded it and in the years that followed. While these contact readings were elevated they remained within design limits predicted for the NUHOMS-24P DSC. It is also important to note that unlike the elevated contact levels recorded, the dose rate measurements at 12 inches from the HSMs grating in question did not vary significantly when compared to similar readings taken in 2000 and 2002.

Based on this information we believe the difference in contact levels is most likely due to monitoring geometry in respect to the HSM grating. The width of the solid portions of the grating is a significant fraction of the width of the active volume of the ion chamber dose rate meter. Recorded contact dose rates are therefore heavily influenced by the position of the active detector volume relative to slots in the grating. If the ion chamber is positioned so as to minimize the amount of shielding provided by the grating, readings will be elevated.

Therefore because the 12 inch readings trend smoothly from year to year, it is then believed the recorded differences in contact dose rates are due to the location of the individual measurements with respect to the solid portions of the HSM grating.

References

1. Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated September 17, 2010, Site-Specific Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application
2. NUREG-1801, Revision 2, Generic Aging Lessons Learned (GALL) Report - Final Report, dated December 2010
3. Letter from Mr. C. H. Cruse (BGE) to Document Control Desk (NRC), dated April 8, 1998, Application for License Renewal
4. NUREG-1536, Revision 1, Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility, Final Report, dated July 2010
5. Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated February 10, 2011, Responses to Request for Supplemental Information, Re: Calvert Cliffs Independent Spent Fuel Storage Installation License Renewal Application

ENCLOSURE 1

RAI 3-1 Tables

Calvert Cliffs Nuclear Power Plant, LLC
June 28, 2011

Enclosure 1 - RAI 3-1 TABLES

TABLE 1-List of "Screened In" Terms

Page	Term	Text	Screen In Criteria	Additional Information
A-1/A2.1	significant	The purpose of the HSM aging management program is to: • Ensure that no significant degradation to the HSMs occurs	Yes-a	Significant degradation means degradation of the HSM to the point where it cannot perform its safety functions.
A-3/A2.1	significant	A set of inspection attributes and acceptance standards for steel and concrete that is commensurate with industry codes, standards, and guidelines has been established. Components are determined to be either Acceptable, Acceptable with Defects, or Unacceptable. Acceptable signifies that a component is free of significant deficiencies or degradation that could lead to the loss of structural integrity. Acceptable with Defects signifies that a component contains deficiencies or degradation but will remain able to perform its design basis function until the next inspection or repair. Unacceptable signifies a component contains deficiencies or degradation that either prevents (or could prevent prior to the next inspection) the ability to perform their design basis function.	Yes-a	Significant means deficiencies or degradation that would prevent the component from performing its safety function.
A-4/A2.2	significant	Operating experience to date has not indicated any significant degradation to any of the HSM components. Inspections and surveillances that would identify any deficiencies continue to be conducted.	Yes-a	Significant means deficiencies or degradation that would prevent the component from performing its safety functions.
A-4 /A2.2	significant	The purpose of the transfer cask aging management program is to ensure that no significant degradation to the transfer cask occurs, with the focus being on the frequently wetted surfaces, as well as carbon steel surfaces, prior to its use for future retrieval of a DSC from the corresponding HSM.	Yes-a	Defined as degradation that would prevent the transfer cask from performing its safety functions.

TABLE 1-List of "Screened In" Terms

Page	**Term**	**Text**	**Screen In Criteria**	**Additional Information**
A-6/A2.3	significant	The purpose of the transfer cask lifting yoke aging management program is to ensure that no significant degradation to the transfer cask lift yoke occurs, with the focus being on the repeatedly and intermittently wetted surfaces prior to its use for movement of the transfer cask.	Yes-a	Defined as degradation that would prevent the transfer cask lifting yoke from performing its safety functions.
A-8/A2.4	significant	The purpose of the cask support platform aging management program is to ensure that no significant degradation occurs while the cask support platform is in the borated water environment of the spent fuel pool.	Yes-a	Defined as degradation that would prevent the cask support platform from performing its safety functions.
A-11/Table A-1	significant	To perform visual inspection of transfer cask exterior surfaces and of carbon steel bolts/washers looking for signs of significant degradation (corrosion).	Yes-a	Defined as corrosion that would prevent the transfer cask and its associated sub-components from performing its safety functions.
A-3/A2.1	large, many, most	A review of industry operating experience identified a large number of events related to dry storage. Many of these were event-driven incidents, and most were not related to aging management.	Yes-a	An OE search on the INPO website for key words: ISFSI, age, crack and failure, which resulted in 300 matches. However only 2 separate events alluded to degradation that may have been associated with age related degradation: 1)OE27586 and OE30767, Millstone roof cracks- These were not attributed to age related degradation but to the fact that concrete was not properly sealed against the elements. 2) OE32809 and OE32365, Peach Bottom seal failure due to environmental degradation - This elastomer component is not on Calvert Cliffs casks. All OEs were reviewed for applicability and none were found that were directly attributable to age degradation.

TABLE 1-List of "Screened In" Terms

Page	Term	Text	Screen In Criteria	Additional Information
A-3/A2.1	minor	Minor cracking was noted on top of the HSMs which required cosmetic crack repair.	Yes-a	Minor defined as superficial cracking that does not endanger the structural integrity of the HSM.
C-4/9.6.5.1, 10955-0101 p. 9 sect 4.1	significant	Since the DSC is filled with the inert helium gas there is no significant corrosion of the DSC shell and other components.	Yes-a	The use of an inert gas prevents the atmospheric conditions necessary that would cause corrosion to occur.
C-4/9.6.5.1, 10955-0101, pg. 9, sect. 4.1	sufficient	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. 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.	Yes-b	The gap for axial thermal expansion can range from 1.11 to 1.41 inches depending on the burnup of the fuel and the tolerance of the inside length of the DSC. This gap allows for unrestrained axial and radial thermal expansion.
10955-0101, pg 13, Sect 4.2	sufficient, low	The storage module is a reinforced concrete structure. The effect of radiation on the HSM concrete is evaluated in Section 8.1.1.5.D and Section 12.8.1.1.5.D of Calvert Cliffs ISFSI USAR [6]. This evaluation demonstrates that the magnitude of the neutron fluence incident on the concrete is low enough to not affect the properties of the concrete. This evaluation also demonstrates that the magnitude of the gamma-ray energy deposition on the concrete is not sufficient to cause any radiation heating in the concrete. Therefore, the thermal analyses documented in [Calvert Cliffs ISFSI USAR [6] implicitly considered the radiation heat effects adequately.	Yes -a	As discussed in ISFSI USAR Sections 8.1.1.5D & 12.8.1.1.5D, the basis for this statement is from Section 8.1.1.5 of NUHOMS-24P Topical Report NUH-02. Topical Report NUH-02 was previously submitted as part of response to request for supplemental information.

TABLE 1-List of "Screened In" Terms

Page	Term	Text	Screen In Criteria	Additional Information
C-6/9.6.5.2, 10955-0101, pg 13, Sect 4.2	significant	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.	Yes-a	Significant means environmental degradation will not impact HSM's ability to perform its safety function since the carbon steel reinforcing steel is not exposed to adverse environmental conditions.
C-7/9.6.5.3, 10955-0101 pg. 15, sect. 4.3	significant	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.	Yes-a	This is a summary statement that is drawn from the information contained in the previous paragraphs.
C-5/9.6.5.2	low	The effect of radiation on the HSM concrete is evaluated in Section 8.1.1.5.D and Section 12.8.1.1.5.D. 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.	Yes-b	The effect of radiation on HSM concrete is discussed in the sections of the USAR cited and in the referenced Topical Report for NUHOMS-24P,NUH-002.
E-6/E2.2	slight	The NUHOMS-32P DSC storage capacity is optimized by reducing the space between the locations of each fuel assembly and by slightly reducing the size of the storage locations.	Yes-b	The center to center spacing of the guide sleeves is 9.125" for the NUHOMS-32P DSC and 10.36" in NUHOMS-24P DSC. The inner diameter of the guide sleeves is 8.5" in the NUHOMS-32P DSC and 8.7" in the NUHOMS-24P DSC.
E-21/E4.1.1	significant	The license amendment for using a NUHOMS-32P DSC (Reference E9.7) confirmed that there is no significant increase in individual or cumulative occupational radiation exposure and that the corresponding dose is bounded by the original Technical Specification limits. Calculations submitted in support of the higher burnup fuel license amendment request (Reference E9.8) confirm that these limits would be maintained for that higher burnup fuel. Section 7.4 of Reference E9.6 provides a detailed discussion.	Yes-c	Term is a summarization of the findings discussed in Section 12.7 and Chapter 7 of Calvert Cliffs IFSI USAR in regards to radiation protection

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
A-10/A2.4	many	Chemistry data for monitored parameters is routinely trended to identify subtle trends in the data which may be indicative of an underlying operational problem. In many cases, this allows correction prior to a parameter becoming out-of-specification.	No-e	Use of the term provides no important information.
C-4/9.6.5.1, 10955-0101, pg. 9, Sect 4.1	significant, less	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 9.15). For the DSC, the neutron fluence (10^{14} neutrons/cm2) is much less than this level for the intended storage period and hence, a TLAA is not required.	No-c	The terms are explained by adjacent quantitative terms of neutron fluence.
C-4/9.6.5.1, 10955-0101, pg. 9, Sect 4.1	low	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 9.15).	No-e	Low -alloy steel is a commonly used term throughout industry.
10955-0101, pg.11, sect. 4.1	most	As seen above, the total number of B^{10} atoms/cm^2 available exceeds the neutron fluence for the total storage period of 60 years assuring the continued efficacy of the neutron absorber materials. This conservatively assumes that all neutrons are absorbed in the B^{10}, however, most neutrons do not get absorbed and contribute to the HSM dose rate during storage.	No-c	Quantifying information is contained in the paragraphs and equations that preceed use of this term.
10955-0101, pg.12, sect. 4.1	less	2) Maximum initial fill gas pressure less than 435 psia and…..	No-e	Less than 435 psia is one of the limitations placed on fuel assemblies considered for storage in the NUHOMS-24P DSC.

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
C-6/9.6.5.2, 10955-0101, pg 13, Sect 4.2	significant	Reference 9.16, 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 9.16)	No-c	The expected number of cycles (21,900) is several magnitudes below the limit (10,000,000) cited in the reference.
C-6/9.6.5.2	significant	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 9.14 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.	No-c	The term is explained by the numercial values provided in this paragraph.
C-7/9.6.5.3	significant	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 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 9.14.	No-c	The term is explained by the numercial values provided in this paragraph.
C-7/9.6.5.3, 10955-0101 pg. 15, sect. 4.3	significant	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 9.14 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.	No-c	The term is explained by the numercial values provided in this paragraph.

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
C-4/9.6.5.1, 10955-0101, pg. 9, sect. 4.1	less	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.	No-c	The heating effects are discussed in the earlier paragraphs of sections (9.6.5.1 and 4.1 respectively) and their associated references.
C-5/9.6.5.1	less	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 9.14	No-c	The term is explained by the numercial values provided in this paragraph and the reference.
E-7/E2.2	small, significant	No gaseous, liquid, or solid wastes are generated at the ISFSI. The small volumes of these wastes, which are generated in the Auxiliary Building, has no significant impact on the ability of existing plant systems to process them and this condition is expected to remain unchanged in the license renewal term.	No-d	Use of these terms is not related to in-scope SSCs.
E-7/E2.2	small	A small amount (<15 ft^3/DSC) of liquid waste results from transfer cask decontamination.	No-c	Amount is quantified (<15 ft^3/DSC).
E-7/E2.2	small	The decontamination procedure results in a small amount of a detergent/demineralized water mixture being collected in the cask washdown pit.	No-d	Term is not related to an in-scope SSC.
E-7/E2.2	small	A small quantity (<2 ft^3/DSC) of low-level solid waste is generated as a result of DSC loading operations and transfer cask decontamination (e.g., disposable Anti-C garments, tape, decon cloths, etc.).	No-c	Amount is quantified (<2 ft3/DSC).
E-9/E3.1	small	There are five small concentrations of population, either cities or unincorporated communities, within 10 miles of the Calvert Cliffs site.	No-e	Use of term is descriptive and does not provide important information.

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
E-9/E3.2	small	Calvert County has one main four-lane road (Maryland State Route 2/4) bisecting the County north to south with smaller roads running from the main road to the water on each side. Very few of the smaller roads off Maryland State Route 2/4 connect with each other; therefore, this highway services the bulk of the traffic for the length of the County.	No-e	Use of term is descriptive and does not provide important information.
E-12/E3.8.1	small	Smaller cities and towns within 50 driving miles include Glenarden, 50 driving miles away, North Beach, 26 driving miles,	No-e	Use of term is descriptive and does not provide important information.
E-25/E5.0	small	Based upon the assessments provided in Section E4.0, the impacts of license renewal and continued operation of the ISFSI are small and would not require additional mitigation.	No-c	This is a summary statement that is drawn from the previous sections of this Environmental Report Supplement
E-26/E8.0	small	All impacts of license renewal of the ISFSI are small and would not require additional mitigation.	No-e	This is a summary statement that is drawn from the previous sections of this Environmental Report Supplement
E-1/E1.0	significant	Based upon this experience, there is significant uncertainty regarding when DOE will be able to take possession of the fuel.	No-d	Term is not quantifiable and is not related to any in-scope SSC or impact aging effects.
E-2/E1.3	significant	In addition, NRC indicates that the supplemental report may be limited to incorporating by reference, updating or supplementing the information previously submitted to reflect any significant environmental change. Therefore, Calvert Cliffs assembled a team to review the Environmental Report submitted with the original ISFSI license application and NRC's subsequent environmental analysis, to identify areas that require updating to meet the expectations of Reference E9.1 and to evaluate whether any significant changes have occurred during the initial licensing period and whether any significant changes are anticipated during the renewal term.	No-d	This paragraph is a re-statement of the words contained in 10CFR51.60 which defines the requirements for the Environmental Report.

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
E-7/E2.2	significant	No construction or refurbishment is currently planned during the ISFSI license renewal term. However, due to the delay in licensing of a federal repository for spent nuclear fuel, the need for long-term spent fuel storage onsite is significant and will require complete build out and loading of the facility (i.e., 120 HSMs).	No-d	Term is not quantifiable and is not related to any in-scope SSC or impact aging effects.
E-7/E2.3	significant	Construction of a new independent spent fuel pool, similar to the existing pool, would involve high commissioning costs and significant operating and maintenance cost, as compared to continued operation of the existing facility.	No-d	Term is not related to an in-scope SSC. Attempting to quantify would add no value.
E-8/E2.3	significant	Another alternative is to license and construct and operate a new ISFSI facility on the Calvert Cliffs site. This would involve environmental impacts from construction, significant cost, and increased personal exposure associated with the transfer of fuel from the existing facility with no environmental benefit over the continued operation of the existing facility.	No-d	Term is not related to an in-scope SSC. Attempting to quantify would add no value.
E-12/E3.8.1	significant	The area surrounding the Calvert Cliffs site has experienced significant growth for the last three and one half decades. The projections for the total population within the 50 mile radius of the site are 29 percent higher than that projected in the original ISFSI Environmental Report for 2010 and 46 percent higher within the 10 mile radius of the site.	No-c	Significant growth is quantified in subsequent sentence.
E-13/E3.8.2	significant	Out-commuting represents a significant change to the population base in the Calvert and St. Mary's Counties. Based on 2000 Census Bureau County-to-County Worker Flow survey data, these counties experienced a net loss of 20,931 persons during the work week/work day/work hour period	No-c	Significant change is quantified in subsequent sentence

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
E-13/E3.8.3.1	significant	• The minority population percentage of the environmental impact area is significantly greater (typically at least 20 percentage points) than the minority population percentage in the geographic area chosen for comparative analysis (in this case the 50 mile geographic area).	No-c	Term is defined as 'typically at least 20 percentage points'
E-14/E3.8.3.1	significant	• The percentage of households below the poverty level in an environmental impact area is significantly greater (typically at least 20 percentage points) than the low-income population percentage in the geographic area chosen for comparative analysis (in this case, the 50 mile geographic area).	No-c	Term is defined as 'typically at least 20 percentage points'
E-22/E4.2	significant, little	As noted, operation, maintenance and surveillance activities would continue to be performed by Calvert Cliffs personnel, and no additional staff would be required to conduct these activities during the license renewal term. Therefore, there would be no significant adverse effect on transportation or regional socioeconomics during the license renewal term. Recent population increases indicate spent fuel storage has not been a significant deterrent to immigration or growth, so continued operations of the facility would have little effect, if any, on land use patterns in the region.	No-d	Terms are not related to an in-scope SSC. All ISFSI related activities during the extension period will continue to be performed by Calvert Cliffs personnel.
E-23/E4.2	significant	Based on the Calvert Cliffs review, the ISFSI license renewal and continued operations would result in no significant impact.	No-e	Summary statement summarizing entire Environmental Report .
E-25/E6.0	significant	• To determine whether any statistically significant increase occurs in the concentration of radionuclides near the ISFSI.	No-d	Overview statement about purpose of REMP program.
E-25/E7.0	significant	The incremental increase in operating costs due to aging management activities is insignificant compared to the construction costs of a new dry fuel storage installation on the current site property or the construction and maintenance costs associated with a new spent fuel pool.	No-d	Term is not related to an in-scope SSC. Attempting to quantify would add no value.

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
E-25/E7.0	significant	A quantitative evaluation of environmental costs of alternatives is not necessary to recognize that significant environmental impacts would be avoided by continued operation of the ISFSI versus the limited available options for storage.	No-d	Term is not related to an in-scope SSC. Attempting to quantify would add no value.
E-26/E7.0	significant	As summarized in the environmental information submitted for the Oconee ISFSI license renewal application, the NRC has extensive experience evaluating environmental impacts from ISFSI facilities, and the NRC has consistently issued findings of no significant impact for the operation of the facilities (Reference E9.12, Section 1.2).	No-d	Term is not related to an in-scope SSC. Statement summarizes the words contained in the cited Reference.
E-21/E4.1.1	low	Calvert Cliffs is committed to a program of keeping occupational radiation exposure as low as reasonably achievable.	No-d	Term is not related to an in-scope SSC. Common industry term.
E-26/E8.2	low	Land required to permanently store or dispose of the spent nuclear fuel and low-level radioactive wastes generated from facility operations	No-d	Term is not related to an in-scope SSC. Common industry term.
E-26/E7.0	minor	The proposed license renewal involves a cost-effective utilization of an existing asset, with relatively minor environmental impact, making it the preferred means of securing long-term spent fuel storage capability.	No-d	Summary statement not related to an in-scope SSC.
E-9/E3.2	few	Very few of the smaller roads off Maryland State Route 2/4 connect with each other; therefore, this highway services the bulk of the traffic for the length of the County.	No-d	Term is not related to an in-scope component.
E-14/E3.8.3.1	few	There are very few concentrations of low-income populations within 50 mile radius of the Calvert Cliffs site. Only 67 census block groups are classified as having low-income populations with 27 in Maryland, 3 in Virginia, 35 in Washington D.C., and 2 in Delaware.	No-d	Term defined in the sentence that follows it.

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
E-7/E2.2	little	The passive nature of the facility requires little maintenance beyond periodic surveillance of the air inlet and outlet vents.	No-	Summary statement meant to show the passive nature of the facility. Other sections in the license renewal provide further information.
E-9/E3.1	little	Other features on note in the vicinity include the Dominion Cove Point Liquified Natural Gas (LNG) import facility (a little over 100 acres of industrial land use) is about 3.5 miles southeast of the Calvert Cliffs site within Calvert County.	No-d	Term is not related to an in-scope SSC.
E-22/E4.2	little	Continued operations of the ISFSI during the 40-year license renewal term would have little to no impact on the following resources:	No-c	Summary statement that is further explained in subsequent paragraphs in Section E4.2.
E-22/E4.1.2	less	The maximum exposed member of the public maintaining continuous occupancy in the nearest residence, which is 4,075 feet from the ISFSI facility, is less than 2 mrem/year when all 120 HSMs are filled to capacity and less than 13.5 mrem/year for the remaining uranium fuel cycle activities in the area.	No-c	use of the tem "less than 2 mrem/hr 'and "less than 13.5 mrem/year" adequately quantify the amount of expected exposure to the public. Futher quantification is not needed.
E-22/E4.1.2	less	The collective dose for persons within 0-2 miles of the ISFSI is conservatively estimated to be 24.3 mrem/year over 2,771 people. This is less than 1 percent of the collective dose from the remaining fuel cycle operations. Therefore, radiation exposure due to the ISFSI, combined with all other fuel cycle operations, would not exceed the regulatory requirements of 25 mrem/year in 10 CFR 72.104 and 40 CFR Part 190.	No-c	The amount of expected dose is quantified in previous sentence. Use of the term "less than 1 percent" adequately quantifies the expected dose in comparision to regulatory requirements.
10955-0101, sect. 2.0,pg. 5,	most	It is important to note, however, that the CCNPP Design Basis Documents for the Calvert Cliffs ISFSI (Specifications for the DSC, HSM and other components) required the facility to be designed for a 50-year service life. Most importantly, the design basis calculations documented in the Topical Report [2] for the NUHOMS® System are based on a service life of 50 years.	No -e	A summary statement where the term does not provide important information

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
10955-0202,sect. 4.0.2, pg 7	significant	S_a - is the value obtained from the applicable design fatigue curve for the total specified number of Significant Pressure Fluctuations. Significant Pressure Fluctuations are those for which the total pressure excursion exceeds the pressure range equal to: 1/3 x design pressure x (S/S_m)	No-c	Significant is defined by the equation contained in this paragraph.
10955-0202,sect. 4.0.2, pg 7	less	S - equals to S_a, determined at 10^6 cycles of applicable ASME code fatigue curve [2.5], if the total specified number of service cycles is 10^6 or less.	No-c	Less refers to any value below $1X10^6$.
10955-0202,sect. 4.0.2, pg 7	significant	According to Reference [2.6], it is assumed that measurable pressure fluctuations, caused by ambient temperature cycles, occur 5 times per year for DSCs. Therefore for the designed operational service life of 60 years, the number of significant pressure cycles is expected to be 600 for DSCs. The corresponding value of S_a associated with this number of cycles is approximately 150 ksi.	No-c	5 occurrences*2 cycle per occurrence* 60 years = 600 cycles
10955-0202,sect. 4.0.3, pg 8	large	A normal operational cycle of startup-shutdown for DSC occurs only once for a whole designed service life. According to Fig. 1-9.2.1 in Ref. [2.5], the value of Sa is very large (>800 ksi), and therefore the value of $S_a/2E\alpha$ is very high (> 1500 °F). This value is far greater than temperature difference between any two points on the DSCs. Therefore, this condition is met for both NUHOMS® 24P DSCs and NUHOMS® 32P DSCs.	No-c	Large is defined as >800 ksia.

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
10955-0202,sect. 4.0.4, pg 8	significant, less	Under this condition, the temperature difference between any two adjacent points shall not change by more than the value of S_a /2E during the normal service, where S_a is the value obtained from applicable fatigue curve for total number of Significant Temperature-Difference Fluctuations. A temperature-difference fluctuation is considered significant if its total algebraic range exceeds the quantity Sa/2Eα, where S is the value of Sa obtained from the applicable fatigue curve at 106 cycles (when the total numbers of thermal cycles equal or less than 106).	No-c	Significant is part of the name for a term described in this paragraph. Less refers to any value below 1X106.
10955-0202,sect. 4.0.4, pg 8	small, less	Small fluctuations of thermal gradients in DSCs during normal storage in HSM occur as a result of seasonal ambient temperature changes [2.6], the ambient temperature cycles causing a measurable thermal gradient fluctuation are assumed to occur 5 times per year [2.6]. Therefore, for an operational life of 60 years, the total number of ambient temperature cycles is 600, which is less than 10^6. Hence, a value of S = 28.2 ksi is taken according to Ref. [2.5].	No-c	Both terms are adequately defined within this paragraph.
10955-0202,sect. 4.0.4, pg 8	most, significant, less	The value of $S_a/2E\alpha$ for the DSCs of NUHOMS® 24P and NUHOMS® 32P is calculated using the material properties from Table 1, which is 54 OF for both NUHOMS® 24P and NUHOMS® 32P. It is shown in Refs. [2.6] that for NUHOMS® 24P the most significant temperature fluctuation occurs during a change in ambient temperature from 0 °F to 100 °F, which results in a change of temperature difference of 32.3 °F. This temperature difference is less than 54 °F. Therefore, this condition is satisfied for NUHOMS® 24P.	No-c	The terms used are adequately defined within this paragraph.
10955-0202,sect. 4.0.4, pg 9	less	Clearly, these temperature fluctuations for both NUHOMS® 24P (32.3 °F) and NUHOMS® 32P (43.1 °F) are less than 54 ° I Therefore, this condition is satisfied.	No-c	Less is any value below 54 °F.

TABLE 2 - List of "Screened Out" Terms

Page	Term	Text	Screen In Criteria	Additional Information
10955-0202,sect. 4.0.4, pg 9	significant, less	The sixth criterion states that specified full range of mechanical loads, shall not result in load stresses whose range exceeds the S_a value obtained from the applicable design fatigue curve for the total specified number of Significant Load Fluctuations. A load fluctuation shall be considered significant only if the total excursion of load stress exceeds the quantity S, where S is defined below: S is the value of Sa corresponding to the 10^6 cycles from the applicable fatigue curve if the total specified number of service cycle is 10^6 or less.	No-c	Meaning of the term Significant Load Fluctuations is defined in the paragraph. Less is a value below $1x10^6$.
10955-0203,sect. 6.0, pg 6-10	significant, less	See pages 6-10 of Areva calculation #10955-0203. Terms are used numerous times throughout the section 6.0.	No-c	Meaning of the term significant pressure fluctuation and significant temperature-difference fluctuations are defined in this section.

ENCLOSURE 2

Table 3.4-1 Revision

Calvert Cliffs Nuclear Power Plant, LLC
June 28, 2011

Table 3.4-1, Aging Management Review Results for the HSM

Subcomponent	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activities
Reinforced Concrete Walls, Roof, and Foundation	HT, SH, SS	Concrete Steel	Yard, Air	Loss of Material Spalling, Scaling and Cracking / Freeze-Thaw[1] Change in Material Properties[1]	ISFSI Aging Management Program
Reinforced Concrete Walls, Roof, and Foundation (Underground)	HT, SH, SS	Concrete Steel	Embedded / Underground	Loss of Material Spalling, Scaling and Cracking / Freeze-Thaw[1] Change in Material Properties[1]	ISFSI Aging Management Program
DSC Structural Steel Support Assembly	SS	Carbon Steel	Sheltered	Loss of Material	ISFSI Aging Management Program
		Nitronic 60 Stainless Steel	Sheltered	None Identified	None Required
DSC Seismic Retainer for HSMs	SS	Carbon Steel	Sheltered	Loss of Material	ISFSI Aging Management Program
Cask Docking Flange and Tie Restraints	SS	Carbon Steel	Sheltered	Loss of Material	ISFSI Aging Management Program
			Yard	Loss of Material	ISFSI Aging Management Program
Heat Shield	HT	Stainless Steel	Sheltered	None Identified	None Required
Shielded Front Access Door and Door Supports	SH, SS	Carbon Steel	Yard	Loss of Material	ISFSI Aging Management Program
		Concrete	Embedded	None Identified	None Required
Ventilation Air Openings (One Inlet / Two Outlets)	HT	Stainless Steel	Yard	None Identified	None Required

Table 3.4-1, Aging Management Review Results for the HSM

Subcomponent	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activities
Shielded Ventilation Air Inlet Plenum (Concrete)	HT	Concrete	Yard	Loss of Material, Spalling, Scaling and Cracking / Freeze-Thaw[1] Change in Material Properties[1]	ISFSI Aging Management Program
Shielded Ventilation Air Inlet Plenum (Stainless Steel)	HT	Stainless Steel	Embedded / Yard	None Identified	None Required
Ventilation Air Outlet Shielding Blocks (Concrete)	HT	Concrete	Yard	Loss of Material, Spalling, Scaling and Cracking / Freeze-Thaw[1] Change in Material Properties[1]	ISFSI Aging Management Program
Ventilation Air Outlet Shielding Blocks (Stainless Steel)	HT	Stainless Steel	Embedded / Yard	None Identified	None Required
Lightning Protection System	SS	Copper/Bronze	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
Ladder and Attachments	None	N/A	N/A	N/A	N/A
Caulk, Sealants, Expansion Joint Fillers	None	N/A	N/A	N/A	N/A
Dry Film Lubricants	None	N/A	N/A	N/A	N/A
Polyvinyl Chloride Drain Pipe	None	N/A	N/A	N/A	N/A
Electrical Conduit, Boxes, and Cable	None	N/A	N/A	N/A	N/A
Alignment Targets	None	N/A	N/A	N/A	N/A

Table 3.4-1 Revision

Notes:

[1] Aging effect conservatively included to meet NRC position for 10 CFR Part 54 plant license renewal (ISG 3).

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