

April 28, 2011

Mr. George H. Gellrich  
Vice President  
Calvert Cliffs Nuclear Power Plant, LLC  
1650 Calvert Cliffs Parkway  
Lusby, MD 20657

SUBJECT: 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)

Dear Mr. Gellrich:

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 to the U.S. Nuclear Regulatory Commission 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. The additional information is identified in the enclosure to this letter. We request that you provide the information by May 27, 2011. Please inform us in writing at your earliest convenience, but no later than May 13, 2011, if you are not able to provide the information by the requested date. To assist us in re-scheduling your review, you should also include a new proposed submittal date and the reasons for the delay. This information was discussed with Mr. Ken Greene of your staff in a conference call on April 21, 2011.

Please reference Docket No. 72-8 and TAC No. L24475 in future correspondence related to this licensing action. If you have any questions, please contact me at (301) 492-3325.

Sincerely,

**/RA/**

John Goshen, P.E., Project Manager  
Licensing Branch  
Division of Spent Fuel Storage and Transportation  
Office of Nuclear Material Safety  
and Safeguards

Docket No.: 72-8  
TAC No.: L24475

Enclosure: Request for Additional Information

cc: CCNPP Service List

Mr. George H. Gellrich  
 Vice President  
 Calvert Cliffs Nuclear Power Plant, LLC  
 1650 Calvert Cliffs Parkway  
 Lusby, MD 20657

**SUBJECT: 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)**

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<b>DATE:</b>	4/13/11	4/21 /11	4/19 /11	4/ 21 /11	4/ 21 /11	4/ 15 /11
<b>OFC:</b>	SFST	SFST	SFST	SFST	SFST	SFST
<b>NAME:</b>	GHornseth	NGarcia-Santos	MRahimi	DPstrak	MWaters	
<b>DATE:</b>	4/19 /11	4/ 21 /11	4/ 22 /11	4/ 27 /11	4/28 /11	4/ /11

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

<b>ACRONYMS</b>	
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

- 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).

- 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).

- 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.

- 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.

- 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.

- 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.

- 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.

- 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.

- 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.

- 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.

- 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.

#### Appendix A: Aging Management Programs

- 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.

- 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.

- A-3 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.

- 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.

- 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.

#### Appendix B: Time-Limited Aging Analysis

- B-1 Provide a description of the expression and assumptions used in Appendix B, 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 B, 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).

- 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).

#### Other

- 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.

- 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 in-scope 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(l) and 10 CFR 72.236(m)

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-cm<sup>3</sup>/sec (American National Standards Institute) (ANSI)-N14.5). Since these leak rates are greater than leaktight criterion ( $1E10^{-7}$  ref-cm<sup>3</sup>/sec, per ANSI-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.

O-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.

Response to Request for Supplemental Information

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.

Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2

cc:

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