George H. Gellrich
Vice President

Calvert Cliffs Nuclear Power Plant, LLC
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657
410.495.5200
410.495.3500 Fax

CENG
a joint venture of
Constellation Energy EDF
CALVERT CLIFFS
NUCLEAR POWER PLANT

December 15, 2011

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Independent Spent Fuel Storage Installation; Docket No. 72-8
Response to Second 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

REFERENCES:

(a) Letter from 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 J. Goshen (NRC) to G. H. Gellrich (CCNPP), dated October 7, 2011, Second 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
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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 15th day of December, 2011.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires: March 14, 2015
Date

GHG/KLG/bjd

Attachment: (1) Second Request for Additional Information for Renewal Application
Enclosures: 1 MCNP5 Analysis File (CD)
2 Calculation CA07718, 2011 Update of ISFSI USAR DSC Leakage Dose Analyses

cc: **(Without Enclosures 1 & 2)**
D. V. Pickett, NRC
W. M. Dean, NRC
Resident Inspector, NRC
S. Gray, DNR

ATTACHMENT (1)

SECOND REQUEST FOR ADDITIONAL INFORMATION FOR RENEWAL APPLICATION

Calvert Cliffs Nuclear Power Plant, LLC
December 15, 2011

ATTACHMENT (1)

SECOND REQUEST FOR ADDITIONAL INFORMATION FOR RENEWAL APPLICATION

By letter dated September 17, 2010, as supplemented February 10, 2011, March 9, 2011, and June 28, 2011, Calvert Cliffs Nuclear Power Plant, LLC, submitted a license renewal application to the U.S. Nuclear Regulatory Commission (NRC) for the Calvert Cliffs 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 (RAI)

Appendix B: Time-Limited Aging Analysis

NRC Question B-1:

Provide further clarification of the calculations presented for estimating the total scalar flux. In particular, provide the mesh tally specifications (e.g. tally cell location, dimensions, energy) used in the MCNP5 model and the relevant model material geometry to interpret the locations of the mesh tally cells in the cask model. Additionally, provide reasons for why there is a large difference in calculated neutron fluxes from the two calculation approaches.

In the RAI response dated June 28, 2011, the licensee provided a partial description of the calculation used in the license renewal application, Appendix B, Section 4.1 for calculation of the estimated total scalar flux. The licensee performed a second calculation to provide an independent check of the approximation used in the license renewal application, Appendix 8, Section 4.1 and determined that the results at the centerline of the active fuel region suggest that the method previously utilized may not reflect the peak neutron flux. In the second calculation, the licensee states that the calculation was performed using an in-house MCNP5 model of a NUHOMS-32P dry shielded canister (DSC) containing the design basis neutron source from Calvert Cliffs Calculation CA0672 1, Section 6.5 (see ADAMS Accession Number ML091680542). The licensee states that the MCNP5 model was based on the transfer cask model used in Calvert Cliffs Calculation CA06750 (see ADAMS Accession Number ML091680544) which was modified to include an explicit model of the basket with homogenized fuel. The licensee reports a new value for the peak neutron flux calculated at the center of the neutron absorber plate based upon a mesh tally from the MCNP5 model; however, the licensee does not provide detailed information on how the mesh tally was implemented. Because there are several assumptions that are made in specifying model mesh tallies (e.g. tally celt location, dimensions, energy) that could potentially affect the calculation of the peak neutron flux, the licensee is requested to provide the specifications for the mesh tallies that were used to calculate the neutron flux and the relevant model material geometry to interpret the locations of the mesh tally cells in the cask model. The licensee is also requested to provide reasons for why the two calculation approaches resulted in large differences in the calculated neutron flux. This information is needed to clarify the license renewal basis for time-limited radiation aging analyses.

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

CCNPP Response B-1:

To clarify the basket dimensions and materials used, the MCNP5 model input (XD89nF) and mesh tally output file utilized to provide the second independent check discussed in Calvert Cliffs' June 28, 2011 responses are included on the enclosed CD (Enclosure 1). The results provided were based on the FMESH14 tally at the 170.5 cm location on the z-axis (middle of the active fuel region). The second check represents a conservative model as fresh fuel was assumed and the absorber plates were modeled as unborated aluminum alloy, and as a result subcritical multiplication was higher than would actually be the case. However, to test the significance of this, a second MCNP5 case [XD89nF2; which is also included on the CD (Enclosure 1)] was run with a NONU card added to turn off fission reactions in the model. While this lowered the total flux in the center of the center plate from 1.69E6 n/cm^{2}s to 9.83E5 n/cm^{2}s (average of values from -3 to +3 on the x-axis), the latter result is still an order of magnitude higher than

the original hand calculation. A more explicit model of the irradiated fuel isotopics would likely yield a result somewhere in between the two MCNP5 calculated values. It is believed that the MCNP5 models yield a higher flux because contributions to flux in the center of the basket from all assemblies is considered, whereas the original hand calculation considers the flux only from a single assembly.

It is important to note however, that even the most conservative of these results continues to support the conclusion that depletion of the boron in the neutron absorber plates would be negligible over a 60 year period, and thus the criticality control function of those plates would not be compromised.

Other

NRC Question O-1:

Provide a description to address assurance that procedures are used to minimize variability in survey measurements to ensure the early detection of degradation of shielding or other release of radioactivity.

In the RAI response dated June 28, 2011, the licensee was requested to justify the radiation survey measurements recorded in 2001 from the CCNPP ISFSI Area Radiation Surveys. In the response, the licensee stated that a review of the radiation survey measurements was conducted and 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. The licensee indicated that the contact readings were elevated but they remained within design limits predicted for the NUHOMS-24P DSC. The licensee also indicated that the dose rate measurements at 12 inches from the horizontal storage modules (HSM) grating in question did not vary significantly when compared to similar readings taken in 2000 and 2002. The licensee stated that based upon this information, they believe the difference in contact levels is most likely due to monitoring geometry in respect to the HSM grating since the recorded contact dose rates are heavily influenced by the position of the active detector volume relative to slots in the grating. The licensee is requested to provide a description of how they will address assuring that procedures are used to minimize variability in survey measurements to ensure the early detection of degradation of shielding or other release of radioactivity.

This is required to evaluate compliance with 10 CFR 72.126.

CCNPP Response O-1:

Routine surveys are performed yearly at the ISFSI storage facility in accordance with Calvert Cliffs procedure for routine radiological surveys. The primary purpose of these surveys is to establish radiological protection requirements for plant personnel, including ensuring hazards in the area are identified with the proper radiological postings. These surveys also ensure that dose rates at the Calvert Cliffs ISFSI remain below the regulatory requirements of 10 CFR 72.104.

The current revision of this procedure only requires one specific set of measurements for ISFSI routine surveys: dose rates at the ISFSI Controlled Area fence. This procedure specifies that dose rates at the fence cannot exceed 500 micro-rem/hr (0.5 mrem/hr). No additional survey points for the ISFSI storage area or HSMs are required by this procedure.

Even though there are no specific HSMs survey points required by this procedure, surveys of the ISFSI facility have been relatively consistent since these surveys were first performed. These surveys consist of contact and 12 inch dose rates taken on the front of the HSMs, typically at the air intake screen where the dose rate is highest (due to there being less shielding at this location than the rest of the front of the HSM).

Routine radiological surveys are subject to variability due to two primary factors: survey technique and instrumentation differences. With the exception of the contact readings taken on the HSM air intake screens in September 2001, readings taken at these locations have been consistent over the years. In addition, the 12 inch readings taken in 2001 are consistent with the survey data from previous and subsequent years.

This is the basis for Calvert Cliffs presumption that instrument geometry, caused by placing a smaller radiation probe (an Eberline E-520 instrument was noted as one of the instruments used for the September 2001 survey) or the angling of the Eberline RO-20 probe at the front vent resulted in the contact readings that were inconsistent with previous or subsequent years.

The Eberline E-520 probe is cylindrical in shape, measuring 1.125 inches in diameter and 6.3 inches long. Due to the probe's small size it could have been placed in a configuration at the HSM front vent where a larger detector would not fit. The Eberline RO-20 also referenced on this survey is 4.2 inches wide by 7.7 inches high. Although much larger than the E-520 probe, the RO-20 could be placed at an angle at the HSM front vent resulting in a higher reading than if the RO-20 probe was placed parallel to the HSM front vent.

Instrumentation differences can also cause variability in survey data. In accordance with American National Standards Institute N323A-1997 (American National Standard Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments) radiation protection instrumentation is calibrated to an accuracy of plus or minus 10 percent. As previously discussed, when instruments with smaller probes such as the E-520 are used, contact dose rates can appear to be higher because the detector can be placed closer to the actual radiation source.

Based on a review of the surveys submitted as part of previous requests for information, Calvert Cliffs believes that the current survey procedures and processes are providing consistent surveys of the ISFSI storage facility given the inherent differences due to survey techniques and instrumentation differences. Early detection of shielding degradation or detection of a release of radioactivity is not compromised by the variability of the survey data. No additional Calvert Cliffs procedural guidance is needed to minimize the variability of these surveys.

NRC Question O-2:

Provide detailed step-by-step calculations to justify the response to the NRC RAI 0-3 presented in the June 28, 2011, letter to the NRC.

The staff has reviewed the response and has determined that further clarification identified below is necessary in order to effectively evaluate the licensee's RAI response. The staff comments are based on review guidance from NUREG 1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities", which contains details on the parameters necessary to perform calculations, such as number of casks, fuel assemblies and rods, total surface area of fuel assemblies and rods, activity, release fractions, source term to dose rate calculation, and X/Q parameters for crud, fines, volatiles, and fission gases. Gas/volatiles compounds treated as particulates should be considered as a fine.

Fines and crud should be included in the analysis considering that lower welds have a leak rate of approximately 1E-3 cc/sec (per the Final Safety Analysis Report (FSAR)). The calculations appear to be based on a leak rate of 1E-4 cc/sec rather than 1E-3 cc/sec, which is the sensitivity of the bubble leak test for bottom, girth, and longitudinal welds. The various leak rates used in the calculations (1 E-3 cc/sec, 1E-4 cc/sec, and 1E-7 cc/sec per RAI Response 0-4) should be noted and explained. The licensee should also provide the calculation for the ·size penetration· (e.g., hole size). Staff calculations show a hole size

much larger than 10 μm for leak rates of 1E-4 cc/sec and 1E-3 cc/sec. Additionally, the basis for the "Cask to HSM" fractions should be stated explicitly. Values of 0.05 and 0.0008 for volatiles, 0 and 0.02 for fines, and 0 and 0.02 for crud do not appear reasonable. Considering the convection heat transfer taking place within the HSM (this convection is the basis for heat removal), the factors should be approximately 1, especially for small particles. Fines and crud must be included in the calculations. Further, the calculations for normal, off-normal and accident conditions should use the release fractions discussed in NUREG-1567.

The licensing confinement basis in this case should be updated to reflect realistically conservative dose estimates, in order to clarify compliance with regulatory dose limits and environmental impacts over the renewal period. The basis should consider a second site dose calculation, considering the effect of the remaining DSCs based on more stringent helium leak rate tests mentioned in the June 28, 2011, RAI Response 0-4.

This is required to evaluate compliance with 10 CFR 72.104, 10 CFR 72.106, and 10 CFR 72.126(d).

CCNPP Response O-2:

Enclosure 2 provides a detailed calculation of normal, off-normal, and accident doses in accordance with NUREG-1567 and provides the clarification requested. Some doses are higher than were reported in Calvert Cliffs' June 28, 2011 response due to changes in release fractions assumed. However all results continue to demonstrate compliance with the limits.

NRC Question O-3:

Clarify the information provided in the response to NRC RAI 0-4 concerning the helium leak test details of the future modules.

The staff specifically requests additional information (testing requirements, standards, etc.) on leak testing the DSC, including the shell, baseplate, lid, port covers, base materials, and all canister welds (lid to shell, base plate to shell, longitudinal, siphon and vent port cover welds, etc.) of the future modules. In addition, clarify what constitutes an "existing module" and provide the helium leak testing requirements for DSC components fabricated during the license renewal period. The staff notes it will evaluate whether the licensing basis for this renewal should be updated to reflect this information, or whether generic actions is warranted for Part 72 licensees outside of the license renewal process.

This is required to evaluate compliance with 10 CFR 72.104,

CCNPP Response O-3:

The term "existing module" refers to the 72 HSMs which are currently installed at Calvert Cliffs ISFSI. Of these 72 HSMs, three currently have not had any spent fuel placed inside them. It is anticipated that we will fill all the remaining HSMs by November 2012. All of the DSCs that will be placed into the remaining empty HSMs will be DSCs of the current approved configuration. These DSCs will undergo helium leak testing using the current process and will be measured against the current Technical Specification criteria.

In our response to RAI O-4 of Reference 1, we committed to changing 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 for any future loading of DSCs beyond the existing modules described above. We have intentionally not discussed the specifics associated with helium leak testing for any DSCs to be used beyond the first 72 as Calvert Cliffs is intending to use a DSC that supports storage of higher burnup spent fuel. The specifics of the DSC and the helium leak testing to be used will be part of the license amendment request submitted seeking their

approval. That license amendment request will provide the information and analysis necessary to allow the NRC to fully evaluate the new DSCs. The license amendment request was submitted on December 8, 2011.

NRC Question O-4:

Provide the visual examination results of the lead cask to be performed in April 2012 to the staff for review prior to re-licensing. These results should include any indications of rust blooms on the cask surface, if present, accompanied by appropriate corrective action.

Rust blooms have been demonstrated to be precursors to stainless steel chloride-induced stress corrosion cracking in environments of interest. Observation of any potential rust blooms could indicate the possibility of stress corrosion cracking on the lead cask.

This information is required to evaluate compliance with 10 CFR 72.120(d).

CCNPP Response O-4:

A copy of the results of the visual examination will be provided to the NRC within 60 days after completion of the visual examination. This will include any findings for rust blooms or other issues as warranted. The lead cask inspection is currently scheduled to begin June 27, 2012.

NRC Question O-5:

Justify that the graphitic dry film lubricant, "Perma·Slik," will not induce galvanic corrosion on the HSM and transfer cask support rails.

Graphite is a noble element on the galvanic scale and will induce corrosion of aluminum and steel in the presence of moisture.

This information is required to evaluate compliance with 10 CFR 72.120(d).

CCNPP Response O-5:

Calvert Cliffs HSM is a totally dry ISFSI storage system, i.e., both internal to the DSC and external to the DSC, including the HSM rails, are considered dry conditions. Therefore, while it is possible that austenitic stainless steel in contact with graphite could be susceptible to galvanic corrosion the requisite electrolyte to provide the required electrical current conduction path is assumed not to be present. The HSM rails which were lubricated prior to DSC insertion with graphite-based Perma Slik RN are not expected to be exposed to moisture via water impingement (from rainfall or other sources) or condensation (from water in the air) where the rails contact the outer surface of the 304 Stainless Steel DSC after insertion into the HSM. The sheltered HSM design prevents water contact and the elevated (above ambient) surface temperature of the DSCs prevents any condensation on the surface of the DSC or the Nitronic 60 support rails. The other requirement for galvanic corrosion is that the water also acts as an electrolyte which occurs through the addition of conductive ions such as found in soluble salts. The possibility of accumulation of salt on the surface of DSCs via a deliquescence process at Calvert Cliffs in the sheltered ISFSI environment is currently under evaluation by Constellation Energy Nuclear Group, LLC, Electric Power Research Institute, Nuclear Energy Institute, and other Utilities for the purpose of determining susceptibility to stress corrosion cracking (SCC). Field inspection data from these studies will also prove beneficial to determining any potential susceptibility to galvanic corrosion at the rails. Only if both appreciable moisture and salt are found at the rail/DSC interface should galvanic corrosion at the rails be a concern and even then the amount of residual graphitic lubricant remaining in contact with the DSC surface after insertion of a DSC into the HSM should exist over a very small area, if present at all.

The transfer cask support rails are typically only exposed to extremely low conductivity deionized (DI) water (Conductivity = 0.06 micro-Siemens/cm) thereby effectively eliminating the possibility for galvanic corrosion. In the event the transfer cask support rails were exposed to borated water they could be susceptible to galvanic effects; however the conductivity of the water is still very low (9.3 micro-Siemens/cm versus "Tap" Water at 500-800 micro-Siemems/cm) such that the ability to conduct current and to cause appreciable galvanic corrosion is greatly inhibited. Graphite-based valve packing is commonly used with stainless steel valve stems and packing glands in borated primary water systems and there is no pressurized water reactor operating experience reporting galvanic corrosion issues at locations where the graphite packing is in contact with a stainless steel stem or packing gland if borated water is the process fluid.

NRC Question O-6:

Provide justification that no welds or cask surfaces currently have or will have a temperature of ≤ 85 °C over the next 40 years, at temperatures bounded by normal conditions of storage. If the surface temperature decreases below 85 °C at conditions in which the stainless steel components are susceptible to stress corrosion cracking, justify why the planned inspections are adequate to detect potential degradation as a result of stress corrosion cracking.

NUREG/CR-7030, "Atmospheric Stress Corrosion Cracking Susceptibility of Welded and Unwelded 304, 304L, and 316L Austenitic Stainless Steels Commonly Used for Dry Cask Storage Containers Exposed to Marine Environments" reports that stress corrosion cracking and pitting corrosion of 304 and 316 stainless steel may occur at temperatures less than 85 °C in bounding relative humidity environments.

This information is required to evaluate compliance with 10 CFR 72.120(d).

CCNPP Response O-6:

Figure 1 below provides the initial and estimated heat loadings of all currently loaded DSCs during the current and extended ISFSI license period. The heat load by assembly in each canister was adjusted from the time of loading for decay to each of the future dates shown using the equation given in Reference 2 (page 132). To provide some sense of when these DSCs will fall below the 85°C temperature threshold discussed in RAI O-6, an estimate of DSC surface temperature was developed. The best information available currently comes from design basis calculations which determine the maximum DSC shell surface temperature (DSC top center) for comparison against design limits. Those calculations give a maximum shell surface temperature of 144°C for a 21.12 kW NUHOMS-32P DSC (Reference 3, page 11) and 121°C for a 15.84 kW NUHOMS-24P DSC (Reference 4, Table 8.1-12), when stored inside an HSM with 70°F (21°C) inlet air temperature. These two maximum temperature points were trended with a nominal 0 kW canister with a 21°C surface temperature. The resulting trend ($T(h) = -0.0947\,h^2+7.8125\,h + 21.111$; where h is DSC heat load in kW) was then used to develop the estimate of the maximum surface temperatures of each DSC by heat load shown in Figure 2 (lower temperatures would be expected at the bottom and ends of the DSC). The results suggest that many of the earliest canisters loaded have spent most of their time in storage below 85°C.

For those DSCs for which the surface temperature has or is projected to fall below 85°C over the next 40 years Calvert Cliffs is investigating the possibility of accumulation of salt on the surface of DSCs via a deliquescence process in the sheltered ISFSI environment. Without such accumulation there is no concern that SCC has occurred. Specifically, Calvert Cliffs has volunteered to be a lead plant for an industry initiative with Electric Power Research Institute, Nuclear Energy Institute, and other utilities for the purpose of determining susceptibility to SCC identified by NUREG/CR-7030. Field inspection data

from this initiative, including a planned inspection of the lead DSC at Calvert Cliffs in 2012, will be used to evaluate the potential for SCC at Calvert Cliffs. The planned visual inspection of the lead DSC will attempt to identify any areas which have experienced SCC pre-cursor surface corrosion (staining or deposits) and quantify the actual chloride concentration on the DSC surfaces. If evidence of surface corrosion and high chloride contents are identified on the lead DSC, additional inspections using surface NDE methods may be required to more fully evaluate the extent of condition, including determining whether SCC is present.

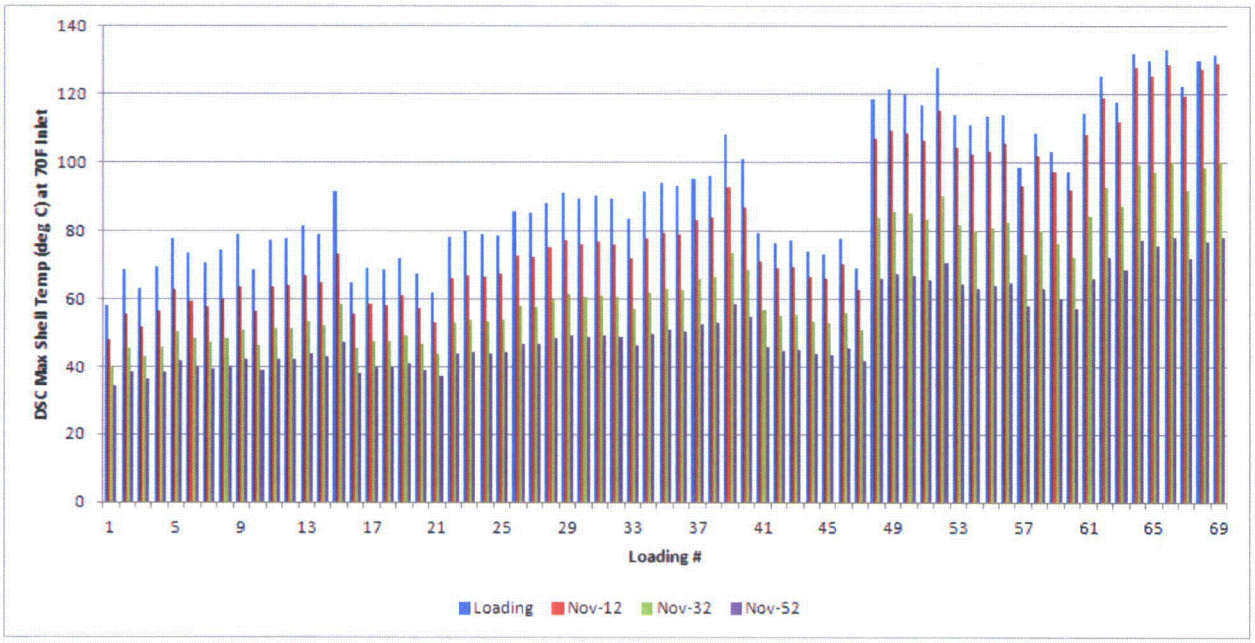

Figure 1 – Heat Loads of Currently Loaded DSCs During the Initial and Extended ISFSI License Period

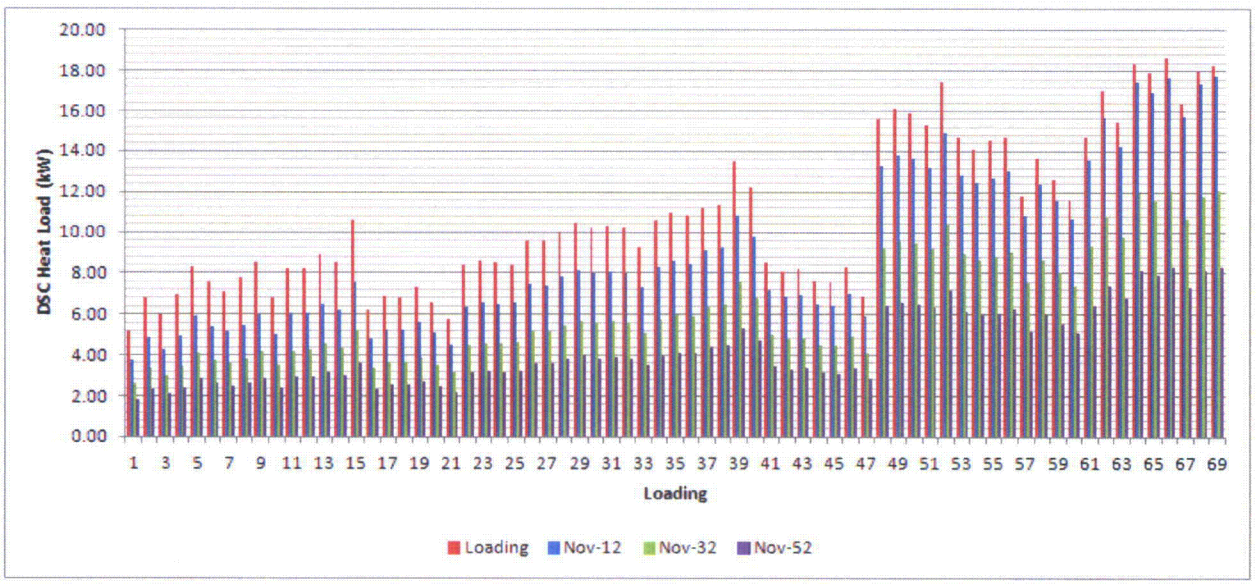

Figure 2 – Estimated Maximum Shell Temperatures of Currently Loaded DSCs During the Initial and Extended ISFSI License Period

NRC Question O-7:

Clarify the industrial codes used for qualification of inspectors, inspection methods, and acceptance criteria for safety-related systems structures and components (SSCs) which require an aging management program (AMP).

Widely recognized industrial codes should be used to govern the inspection and maintenance of aging management procedures, (e.g., American Concrete Institute codes for the AMP of concrete).

This information is required to evaluate compliance with 10 CFR 72.120(d).

CCNPP Response O-7:

The industrial codes used for qualification of inspectors, inspection methods, and acceptance criteria for safety-related systems, structures, and components which require an Aging Management Program are:

- American Concrete Institute 201.1R
- American Concrete Institute 349.3R

NRC Question O-8:

Provide the schedule date for the lead cask inspection to the staff when the date is finalized. (Currently scheduled in April 2012)

The schedule is required for NRC staff to observe the inspection.

CCNPP Response O-8:

Due to an extended refueling outage planned for Calvert Cliffs in first quarter 2012, the canister inspection has been currently rescheduled for June 27, 2012. Calvert Cliffs will keep the NRC informed of any schedule changes in advance to facilitate their schedule.

REFERENCES

1. Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated June 28, 2011, Responses to Request for Additional Information, RE: Calvert Cliffs Independent Spent Fuel Storage Installation License Renewal Application
2. Calculation CA06721, Source Terms for ISFSI 32P Burnup Extension, dated March 13, 2007 (ADAMS ML091680542)
3. Calculation CA06301, Transnuclear Calculation 1095-18, NUHOMS-32P HSM Thermal Analysis – Normal Storage Conditions, dated August 25, 2003
4. Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P, Nutech Engineers, Inc., NUH-002, Revision 1A, July 1989

ENCLOSURE 1

MCNP5 Analysis File (CD)

Calvert Cliffs Nuclear Power Plant, LLC
December 15, 2011