December 22, 2006

Mr. James A. Spina, Vice President
Calvert Cliffs Nuclear Power Plant, Inc.
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING IMPLEMENTATION OF ALTERNATIVE SOURCE TERM (TAC NOS. MC8845 AND MC8846)

Dear Mr. Spina:

By letter dated November 3, 2005, Calvert Cliffs Nuclear Power Plant, Inc. (the licensee) requested an amendment to revise its accident source term in the design basis radiological consequence analyses at Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2.

The Nuclear Regulatory Commission (NRC) staff has reviewed the information provided and has determined that additional information is needed to complete its review. Enclosed is the NRC staff's request for additional information (RAI). This RAI was discussed with your staff on December 18, 2006, and it was agreed that your response would be provided within 90 days from the date of this letter.

If you have any questions, please contact me at 301-415-1457.

Sincerely,

/RA/

Patrick D. Milano, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosure:
RAI

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

IMPLEMENTATION OF ALTERNATIVE SOURCE TERM

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-317 AND 50-318

By letter dated November 3, 2005 (Agencywide Documents Access and Management System Accession No. ML053200316), Calvert Cliffs Nuclear Power Plant, Inc. (the licensee), requested an amendment to revise its accident source term in the design basis radiological consequence analyses at Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. In order to complete its review, the Nuclear Regulatory Commission (NRC) staff requests the following additional information.

ACCIDENT AND TRANSIENT ANALYSES

Steam Generator (SG) Tube Rupture (SGTR) Accident

1. Are the SG blowdown system and waste processing system safety-related systems? If not, what is the basis for assuming credit for these systems in the mitigation of SGTR accident activity releases? Are these systems part of the TMI Action Item III D.1.1 Leakage Reduction Program, and was their leakage incorporated in the analyses of post-SGTR accident activity releases? If not, why?

2. How are tank ventilation and other possible activity leakage pathways from the SG blowdown and waste processing systems accounted for in taking credit for their mitigative function?

3. What is the pathway “Iodine via ADV [atmospheric dump valve] of Unaffected SG – Isolated after 30 Days” used to model, in relation to the design-basis accident (DBA).

4. Clearly identify the coolant mass flow rate for releases from the tube break and ADV steaming for both the faulted and intact SGs.

5. On page 20 of Enclosure 4 of the November 3, 2005, application, it is indicated that steaming is from the main steam safety valves (MSSVs). However, elsewhere in the analysis the steaming release is described as flowing from the ADVs. Clarify the reason for this apparent difference/discrepancy. If the release points are in fact being correctly described, verify and explain how the correct control room atmospheric dispersion factors are being used.

Seized Rotor Event (SRE)

1. In the SRE analysis, it appears that the fraction of moisture carryover was applied only to alkaline metals released from the failed fuel. If this is indeed the case, provide the justification for not applying this fraction to other particulate nuclides and iodine, in particular.

2. As stated in the submitted SRE Design Analysis CA06451, Revision 0, as a design basis, it is assumed that an 8-hour period is required to establish shutdown cooling (SDC) from a hot full power (HFP) condition, which is consistent with "actual plant operation." After the 8-hour period wherein SDC is established, the postulated RCS activity release to the environment ceases. Because of this dynamic activity release, it cannot be assumed that the activity concentration in the environment, resulting from this release, is at its highest from the onset of the accident-initiated leakage; moreover, and more specifically, the determination of a worst 2-hour Exclusion Area Boundary (EAB) dose must account for the actual activity release profile. It appears that the licensee attempted to compensate for this effect by postulating an activity release case that assumes SDC is restored in 2 hours, as opposed to 8. However, this case misses the peak environmental activity concentration associated with the design basis 8-hour release period, which may in fact be even higher than that associated with the hypothetical 2-hour release period case. By assigning the EAB dispersion coefficient from time equal to 0 through the end of the accident, in the 8-hour period case, the code would be allowed to identify the peak 2-hour dose period associated with the DBA release scenario.

   Verify that the methodology used to determine the worst-case 2-hour EAB dose for the design-basis SRE scenario, as described in the application, was both conservative and accurate.

3. Describe (preferably by showing equations) the derivation of the individual release fractions input to the RFT files that are shown in Attachments H through M of Design Analysis No. CA06451, Revision 0.

4. Regulatory Guide (RG) 1.183, Appendix H, can be interpreted as assuming that all fuel reaching fuel melt initiation temperature completely melts and is released in accordance with the specified release fractions. Verify that the licensee's assumption of 5% fuel failure for the SRE conservatively addresses this interpretation.

<u>Main Steamline Break (MSLB) Accident</u>

1. Provide additional justification for the postulation that the main steam piping room is the worst-case, credible, MSLB accident activity release point. In this justification, include a discussion about the validity of assuming a vent release, as opposed to leakage (caused by differential pressure), from this main steam piping room.

2. Provide the basis for the reactor cooldown time assumed for this accident analysis.

3. For the MSLB accident-initiated, concurrent, iodine release rate spike, RG 1.183, Appendix E, recommends an assumed spike duration of 8 hours. Explain the reason why a 9-hour accident-initiated, concurrent, iodine release rate spike duration was assumed in the licensee's MSLB accident analysis.

Fuel-Handling Accident (FHA)

1. In paragraph 4 on page 8 of Design Analysis No. CA06450, Revision 0, it is indicated that a decontamination factor (DF) of 200 is used for FHA analysis cases that assume a drop over seated assemblies in the spent fuel pool (SFP), despite a Technical Specification (TS) requiring only 21.5 feet of water coverage over seated assemblies. The rationale given for using a DF of 200 relies on an assumption that an assembly drop over the SFP will strike the bottom of the pool, as opposed to the top of the racks, thereby maintaining more than the 23 feet of water coverage that is typically required to assume a DF of 200. The result of this analysis ultimately shows that the FHA over the SFP during reconstitution/inspection, and not over seated assemblies, is the limiting FHA case. As explained, this is because a DF of 120, associated with a conservatively assumed value of 20.5 feet of water coverage, is used for analyzing an FHA over the SFP during reconstitution/inspection. However, there is no verification of, or justification for, the assumption that a dropped assembly must strike the bottom of the SFP before resulting in an FHA for the case assuming only seated assemblies. Therefore if, in the design basis FHA analysis, the licensee intends to continue to include the assessments that assume a DF of 200 for an FHA resulting from a drop over seated assemblies in the SFP, provide verification that the dropped assembly will strike the bottom of the SFP and indeed be covered by at least 23 feet of water.

Control Element Assembly Ejection (CEAE) Accident

1. In the CEAE Design Analysis CA06454, Revision 0, as a design basis, it is assumed that an 8-hour period is required to establish shutdown cooling (SDC) from a hot full power (HFP) condition, which is consistent with "actual plant operation." After the 8-hour period where SDC is established, the postulated reactor coolant system (RCS) activity release to the environment ceases. Because of this dynamic activity release, it cannot be assumed that the activity concentration in the environment, resulting from this release, is at its highest from the onset of the accident-initiated leakage; moreover, and more specifically, the determination of a worst 2-hour EAB dose must account for the actual activity release profile. It appears that the licensee attempted to compensate for this effect by postulating an activity release case that assumes SDC is restored in 2 hours, as opposed to 8. However, this case misses the peak environmental activity concentration associated with the design basis 8-hour release period, which may in fact be even higher than that associated with the hypothetical 2-hour release period case. By assigning the EAB dispersion coefficient from time equal to 0 through the end of the accident, in the 8-hour period case, the code would be allowed to identify the peak 2-hour dose period associated with the DBA release scenario.

   Provide verification that the methodology used to determine the worst-case 2-hour EAB dose for the design basis CEAE accident scenario, as described in the submittal was both conservative and accurate.

2. It appears that the CEAE accident analysis only accounts for releases of activity associated with iodine and noble gas isotopes. However, in accident scenarios where fuel failure is postulated, other particulate nuclides are available for release from the gap of failed fuel, as well as from the melted fuel itself. RG 1.183, Position 3, gives guidance to indicate that at least fractions of the "Alkali Metal" isotopic inventory should

be accounted for when fuel failure is postulated for non-loss-of-coolant accident (LOCA) events. Table 3 of RG 1.183 specifies these fractions. This guidance appears to have been followed in the SRE analysis. Therefore, explain why not accounting for the release of these isotopes, in the case of the CEAE accident, is conservative and accurate. Also, indicate how RG 1.183, Footnote 11, is addressed in the analysis of this accident.

3. RG 1.183, Appendix H, can be interpreted as assuming that all fuel that reaches fuel melt initiation temperature, completely melts and is released in accordance with the specified release fractions. Explain how the assumption of 8% melted fuel and 2% clad damage used for the CEAE accident addresses this interpretation.

4. Clarify (preferably by showing equations) the derivation of the individual release fractions input to the RFT files, shown in Attachments H through M of Design Analysis CA06454, Revision 0.

5. RG 1.183, Appendix H, Section 1, indicates that, for airborne activity releases from the primary system following a CEAE accident, an assumption of 25% of the melted fuel fraction of iodine is available for release from containment. One would typically assume that the 25% value is meant to take credit for plateout associated with the release to containment resulting from this particular CEAE accident pathway. Where no containment leakage pathway is postulated, a fraction of 50% is typically used, as can be seen for the boiling-water reactor control room emergency air guidance in Attachment C of RG 1.183. Therefore, provide information to justify the use of Powers' Containment Natural Deposition Model for crediting iodine removal in the CEAE, reactor pressure vessel (RPV) breech case, while also assuming a 25% (as opposed to 50%) iodine activity release from containment through this pathway.

<u>Maximum Hypothetical Accident (MHA)</u>

1. In the MHA Design Analysis CA06449, Revision 0, as a design basis, it is assumed that an 8-hour period is required to establish SDC from a HFP condition, which is consistent with "actual plant operation." After the 8-hour period where SDC is established, the postulated RCS activity release to the environment ceases. Because of this dynamic activity release, it cannot be assumed that the activity concentration in the environment, resulting from this release, is at its highest from the onset of the accident-initiated leakage; moreover, and more specifically, the determination of a worst 2-hour EAB dose must account for the actual activity release profile. It appears that the licensee attempted to compensate for this effect by postulating an activity release case that assumes SDC is restored in 2 hours, as opposed to 8. However, this case misses the peak environmental activity concentration associated with the design basis 8-hour release period, which may in fact be even higher than that associated with the hypothetical 2-hour release period case. By assigning the EAB dispersion coefficient from time equal to 0 through the end of the accident, in the 8-hour period case, the code would be allowed to identify the peak 2-hour dose period associated with the DBA release scenario.

Please provide verification that the methodology used to determine the worst-case 2-hour EAB dose for the design basis MHA scenario, as described in the submittal, was both conservative and accurate.

2. It appears that the licensee assumes an instantaneous release of core activity for the "early in-vessel" phase, as opposed to the 1.3 hour, linear, release specified in the guidance of RG 1.183. Due to the credit being taken for time-dependent containment removal mechanisms, it is possible that the instantaneous methodology implemented may yield comparatively non-conservative results. Additionally, this instantaneous early in-vessel phase release methodology will likely skew the determination of the worst-case 2-hour EAB dose. Therefore, provide verification that the instantaneous early in-vessel phase release methodology that has been used is conservative when compared to the regulatory guidance.

3. Were both trains of the penetration room emergency ventilation system (PREVS) assumed in the analysis to be in operation? If so, verify that both trains will be available following the loss of offsite power associated with the DBA analysis.

4. Do the licensee's procedures allow for the refueling water tank (RWT) to be refilled after an accident? If so, provide verification that this was accounted for in the analysis of MHA RWT release pathway, i.e., was the motive force to flushing the RWT considered along with pH considerations?

5. Clarify and provide additional details as to the RWT release methodology that is implemented. Such additional details may be included in the text of Reference 34 of Design Analysis CA06449, Revision 0.

6. Clarify how PREVS is associated with the hydrogen (H_2) purge line leakage.

7. Provide additional details as to how the H_2 purge line leakrate was determined, and verify that there are no other driving forces (i.e., operating fans, etc.) associated with a post-LOCA release through the H_2 purge lines.

8. The treatment of all isotopes (excluding noble gases) as particulate conservatively results in the accumulation of all relevant activity, regardless of chemical form, being accumulated in one location. However, by assuming 100% particulate, a resultant assumption is made that all activity is depositing on the human-efficiency particulate air filter of the train, <u>not</u> the carbon filter. This may or may not be conservative, depending on the location of the human error probability filter, versus the carbon filter, in relation to the assumed control room operator position and other dose receiver locations. Additionally, effects such as oblique angle scattering, which are not well modeled by the point-kernel method (Micro Shield) being used, may lead to unpredictable dose contributions that are dependent on the geometry caused by different filter locations.

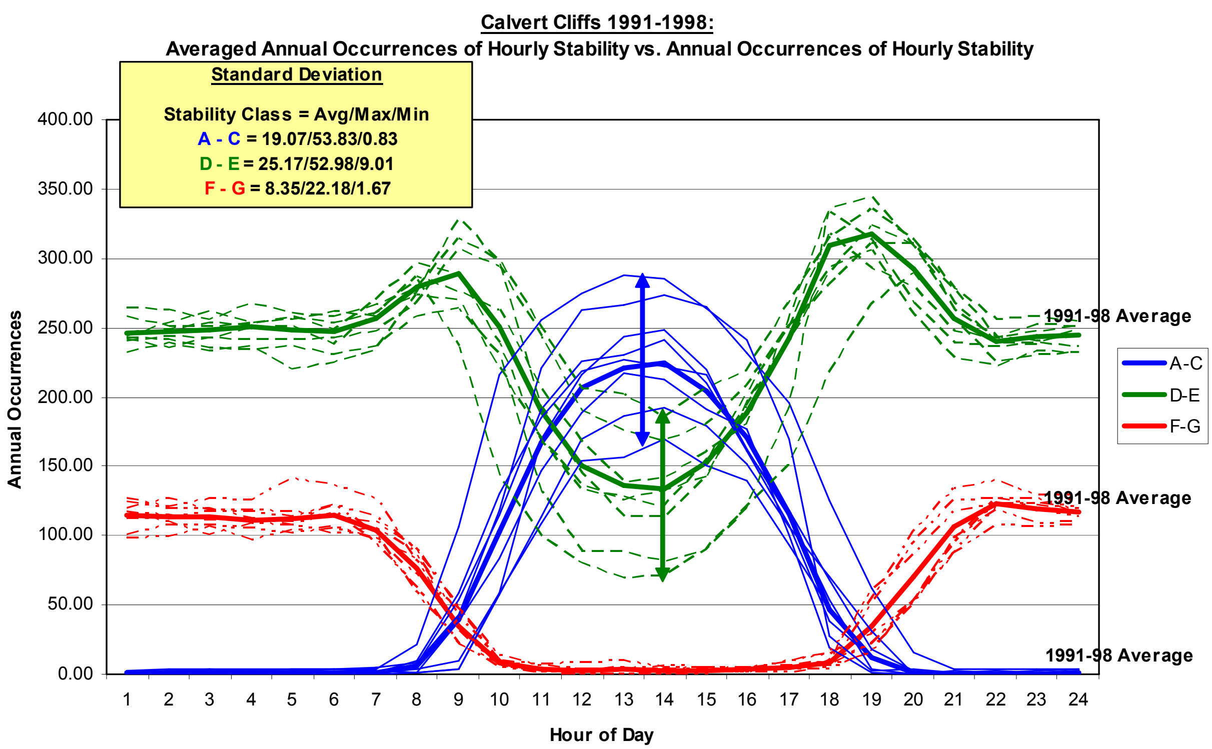
   Therefore, if the licensee chooses to continue modeling all activity, regardless of chemical form, in the same filter location, provide verification that the filter locations selected are limiting with regard to dose, and well modeled by a point-kernel methodology.

9. Provide verification that the larger dimensions that characterize a 10,000 cfm filter will not bound those of the smaller 2000 cfm filter being assumed, despite the associated reduction in source concentration.

Waste Processing Incident (WPI) and Waste Gas Incident (WGI)

1. Explain the rationale for re-analyzing these two accidents, as neither are required for full-scope AST implementation.

METEOROLOGICAL DATA



The NRC staff performed confirmatory analysis of the hourly atmospheric stability values provided and noticed some variations relative to the distributions for certain classes. If the stability classes A thru G are combined to form only three different class groups: Unstable (A-C), Neutral (D-E), and Stable (F-G), the variance becomes apparent. The above figure is a composite view of each year's hourly stability annual occurrences and their variation from the annual mean for all 8 years combined (1991 - 1998):

1. In the figure, it shows that unstable stability classes A-C have an average standard deviation of 19.07, a maximum of 53.83, and a minimum of 0.83. Neutral stability classes D-E have an average of 25.17, a maximum of 52.98, and a minimum of 9.01. Stable stability classes F-G were the most behaved data with an average standard

deviation of 8.35, a maximum of 22.18, and a minimum of 1.67. Given the uncertainty in the data, justify why annual calculations of atmospheric dispersion factors (χ/Q values) should not be performed independently (each year separately) providing the most conservative resulting values listed for accidental atmospheric releases.

ATMOSPHERIC DISPERSION FACTORS

1. Confirm that the cross-sectional area (A) used in the EAB χ/Q calculation for each accident scenario (MHA, FHA, MSLB, SGTR, SRE, CEAEA, WGI, and WPI) is in fact “0 m” as denoted in Enclosures 1-8 of the application. It was noted in Calvert Cliffs Updated Final Safety Analysis Report (UFSAR), Section 2.3.6.1, that the EAB 0-2 hour relative concentration value was calculated using a wake factor (c x A) of 820 m^2 (0.5 x 1640 m^2). Justify the reasoning for changing this value to 0 meters for sources: ADVs, ventilation stacks (VSs), main steam goosenecks (MSGs), RWTs, and containment outage doors (CODs). Note that the value of “A” is defined as the vertical-plane cross-sectional area of the reactor building, measured in m^2 (refer to RG 1.145, Section C - 1.3.1).

2. The NRC staff routinely performs confirmatory analysis on data presented by a licensee when submitting a change in the current licensing basis. In order to conduct these evaluations for the low-population zone (LPZ) atmospheric dispersion estimates (χ/Q values), provide the χ/Q values for the time intervals of 2 - 8 hours, 8 - 24 hours, 24 - 96 hours, and 96 - 720 hours. Accompany these values with any inputs and assumptions made to validate that the most conservative offsite atmospheric dispersion estimate was used.

3. In reviewing the source/receptor distances (Attachment C of Enclosure 9, Atmospheric Dispersion Coefficient (χ/Q) Calculation CA06012, Rev. 0) for the eight postulated accidents, the methodology shown uses mostly straight-line calculations and taut-string calculations for the containment unit source releases. However, reviewing the plant layout and relative distances raised some concern for the length between the Access Control 13 (AC13) and north wall of the Auxiliary Building. If drawn to scale, the layout shows that the distance D, listed as 16.3031 feet on page 34, more accurately resembles 25.6667 feet, listed as distance E on page 34, and vice-versa. If this is so, all source/receptor distance calculations performed involving AC13 are inaccurate and need to be corrected with the appropriate distance of 25.6667 feet for D and 16.3031 feet for E (relative to page 34 of Attachment C). Note that these values are used throughout the source/receptor calculations in Attachment C. Accordingly, all ARCON96 χ/Q calculations involving AC13 will need to be reevaluated using these corrected distance values for the source/receptor points ADV1AC13, ADV2AC13, COD1AC13, COD2AC13, CTMT1AC13, CTMT2AC13, MSG1AC13, MSG2AC13, RWT1AC13, RWT2AC13, VS1AC13, AND VS2AC13. Evaluate these concerns.

4. Provide a copy of the input and output files used to calculate χ/Q EAB and LPZ values for the PAVAN computer code.

VENTILATION SYSTEMS

1. With respect to deleting TS 3.7.10, “Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (PREF)”, if this TS is deleted and design-basis accident doses are maintained within limits, discuss the requirements and the means by which the system will continue to establish and maintain (i.e., drawdown and maintain) a negative pressure so that materials that is released will be release in a manner consistent with the accident analysis. The discussion should include justification for not having a TS for drawdown and a surveillance requirement (SR) to maintain the system. The NRC staff believes that TS 3.7.10 needs an SR somewhat similar to SR 3.7.11.3, which demonstrates that the ECCS pump room area is maintained at a negative pressure (e.g., 0.25 inches water gage with respect to adjacent areas).

2. The definition of L_a, as proposed in Attachment 1 of the November 3, 2005, application, is different than the markup of the TS in Attachment 2 of that submittal. In Attachment 1 the requested change is, “L_a is reduce from 0.20 percent of containment air weight per day at P_a to 0.16 percent of containment air volume per day at P_a.” The marked-up TS in Attachment 2 reduces the maximum allowable containment leakage rate, L_a, from 0.20 to 0.16 percent of containment air weight per day at P_a. It is not clear what is being requested. Clarify this aspect of the request.

3. Within the Ventilation Filter Testing Program (VFTP), the ECCS PREFS and SFPEVS acceptance criteria should have a flow rate. There should also be an SR that links the system’s flow rate to the demonstration of a negative pressure with respect to adjacent areas. Provide this information.

4. It appears that the acceptability of the control room operator doses is dependent upon the installation of the automatic isolation dampers and radiation monitors at the Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof. The licensee should provide a TS and an SR that incorporates the isolation of these dampers on a high radiation signal. The requirement should be for all modes of operation as credit is taken for this function during an FHA.

5. Calvert Cliff’s control room envelope inleakage number is based upon a system configuration and operating characteristics that is no longer relevant. The inleakage test was performed with a CREFS recirculating rate of 2,000 cfm. The new system will be recirculating 10,000 cfm. An inleakage test (e.g., ASTM E741) should be performed based upon the new CREFS recirculating flow rate to confirm that the inleakage is less than the 3,500 cfm assumed in accident analyses.

6. Each of the accident analyses assumed that the CREFS will begin operation 20 minutes following the beginning of the accident. There is no discussion of the operational status of the control room ventilation systems during this 20-minute period. Provide a discussion of the configuration of the CREFS during this period and determine the control room envelope inleakage using an acceptable test protocol (e.g., ASTM E741) to confirm that the inleakage is less than the 3,500 cfm assumed in the accident analyses.

7. With regards to the new CREVS operating scheme and damper isolation functions, has the licensee confirmed that sufficient radioactivity is released to cause isolation at the Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof? If not, does the normal system continue to operate? If it does, what is the inleakage rate in this mode and what are the dose consequences to the control room operatiors?

8. The inplace test criteria in the VFTP for all ESF ventilation system charcoal adsorbers and HEPA filters is 1%. Confirm that all of the accident analyses account for this 1% bypass of the adsorber and HEPA filter.

9. Because the flow rate for the CREVS has changed from 2,000 to 10,000, discuss the change in the residence time during emergency operation and when testing the carbon adsorber, and confirm that the residence time during testing is the same as it is during emergency operation.