January 4, 2005

Mr. George Vanderheyden, 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 (RAI) RE: LICENSE AMENDMENT REQUEST TO USE ALTERNATE DECAY HEAT REMOVAL IN MODE 6 REFUELING (TAC NOS. MC3595 AND MC3596)

Dear Mr. Vanderheyden:

By letter dated June 7, 2004, you requested an amendment to incorporate the use of an alternate cooling method to function as a path for decay heat removal when in Mode 6 with the refueling pool fully flooded. In reviewing your submittal, the Nuclear Regulatory Commission (NRC) staff has determined that additional information contained in the enclosure to this letter is needed to complete its review. An advanced copy of these questions was provided and discussed with your staff on November 10, 2004. The NRC staff had also requested information for the subject license amendment request in a letter dated October 27, 2004. As discussed with your staff, we request you respond to both RAIs within 60 days of the date of this letter to support our review schedule.

If you have any questions, please contact me at 301-415-1030.

Sincerely,

/RA/

Richard V. Guzman, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosure: As stated

cc w/encl: See next page

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ACCESSION NUMBER: ML050050011

OFFICE	PDI-1/PM	PDI-1/LA	SPLB/SC	SRXB/SC	PDI-1/SC
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DATE	12/20/04	12/20/04	1/4/05	1/4/05	1/4/05

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REQUEST FOR ADDITIONAL INFORMATION (RAI) CONCERNING

AMENDMENT REQUEST TO USE

ALTERNATE COOLING METHOD IN MODE 6 REFUELING

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

DOCKET NOS. 50-317 AND 50-318

By letter dated June 7, 2004, Calvert Cliffs Nuclear Power Plant, Inc. (CCNPPI), requested a license amendment to revise Technical Specification (TS) 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation-High Water Level," to permit the operation of a spent fuel pool (SFP) cooling loop providing cooling to the refueling pool in place of one SDC loop operable and in operation. This TS applies in Mode 6, Refueling, with 23 feet or more water above the top of irradiated fuel seated in the reactor vessel. Existing TS 3.9.3, "Shutdown Cooling (SDC) and Coolant Circulation-Low Water Level," permits one SFP cooling loop capable of providing adequate cooling to the refueling pool and reactor vessel to replace one of two SDC loops required to be operable in Mode 6 with less than 23 feet of water above the top of irradiated fuel seated in the reactor vessel.

The provision allowing the substitution of one SFP cooling loop for one of two operable SDC loops in Mode 6 at low water level was issued by Amendment Nos. 55 and 38 for Unit Nos. 1 and 2, respectively, dated June 16, 1981. The preceding amendment request was based on a Nuclear Regulatory Commission (NRC) Generic Letter (GL) addressed to all operating pressurized-water reactors (later numbered GL 80-053) dated June 11, 1980, that requested each licensee to propose an amendment to their facility's TSs to improve the reliability of reactor decay heat removal (DHR) capability. The licensee for CCNPP 1 and 2 proposed the model TSs included with GL 80-053, with modifications including the provision to substitute a spent fuel cooling loop for one of two operable SDC loops. The NRC approved the amendment on the basis that the revised TS was more conservative than the existing specification in that it required an additional DHR subsystem be maintained operable in the event the operating subsystem became inoperable. The amendment did not include an evaluation of the reliability or capability of the SFP cooling loop in providing DHR for irradiated fuel located in the reactor vessel.

By letter dated May 12, 2003, Westinghouse Electric Company submitted Topical Report WCAP-15782, Rev. 0 (non-proprietary), "Use of Alternate Decay Heat Removal in Mode 6 Refueling," for NRC review. The report included a description of the alternate DHR system alignment, a description of system modeling, a description of validation of model predictions to in-plant measurements, a qualitative assessment of risk during operation of alternate DHR, a description of a method to determine alternate DHR entry conditions, and a proposed change to TSs. This topical report relied, in part, on plant-specific information from Calvert Cliffs Nuclear Power Plant, including details used in the qualitative assessment of risk and the validation of system modeling. The license amendment request dated June 7, 2004, references this topical report.

In order to complete our review, the NRC staff requests that CCNPPI provide the following information:

Enclosure

1. **Pressure Boundary Qualification:** Regulatory Guide (RG) 1.29 and Standard Review Plan (SRP) Section 5.4.7 specify that components whose failure could cause an unacceptable reduction in the capability of the residual heat removal system should be safety related (i.e., designed as Seismic Category I and subject to the quality assurance criteria of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B). With the plant in the configuration permitted by the proposed revision to TS 3.9.4, a significant loss of cooling water inventory would cause an unacceptable reduction in the capability of the residual heat removal system in that both the alternate DHR path and the large capacity heat sink provided by the coolant inventory would be lost. SRP Section 3.2.2 and position C.1.b of RG 1.26 specify that portions of the shutdown cooling system essential for performance of the residual heat removal safety function be designed to Quality Group B. Describe the extent that the following pressure boundaries necessary to prevent a significant loss of coolant inventory from the refueling cavity satisfy the criteria of position C.1 of RG 1.29 and position C.1.b of RG 1.26: the refueling cavity (e.g., the refueling cavity-to-reactor vessel seal. interfacing piping, and the cavity liner), the fuel transfer canal (assuming the fuel transfer tube is open), any regularly used temporary reactor coolant system boundaries (e.g., steam generator nozzle dams), and the portions of the SFP cooling loop connected to the refueling cavity.

2. **Loss of Inventory Events and Mitigation Capability:** Section 3.1.3 of WCAP-15872 includes the statement that, for events such as a reactor cavity seal failure or a steam generator nozzle dam failure that cause a significant loss of cooling water inventory, the traditional SDC loop should continue to function while the alternate DHR capability would be unavailable. Industry operating experience includes examples where the pressure boundary formed by the reactor cavity seal and steam generator nozzle dams have failed such that a rapid and gross loss of cooling water did result or could have resulted. NRC SRP Section 5.4.7 and General Design Criterion (GDC) 34 specify that the residual heat removal function be available assuming a single failure. For pressure boundary components such as refueling cavity seals and steam generator nozzle dams that do not meet the quality criteria specified in position C.1 of RG 1.29 and position C.1.b of RG 1.26, explain how the loss of the available alternate DHR path following the failure of these components would be prevented or mitigated to retain the residual heat removal function. If necessary, describe the basis for excluding the systems, structures, and components that provide this mitigation capability from the TSs considering the requirements of 10 CFR 50.36.

3. **Containment closure:** The existing action statements associated with TS 3.9.4 require that containment closure be achieved within 4 hours of a loss of forced cooling for the DHR function. The basis for the 4-hour completion time is the low probability of boiling beginning within the 4-hour period. In the alternate DHR configuration, a steam generator nozzle dam failure could result in reactor vessel boiling in a short time because no SDC loop would be readily available to cool the small inventory of water that would remain in the reactor vessel below the elevation of the steam generator manway. Time to boil would be longer for a refueling cavity seal failure because the remaining cooling water inventory is slightly larger. For pressure boundary components such as refueling cavity seals

and steam generator nozzle dams that do not meet the quality criteria specified in position C.1 of RG 1.29 and position C.1.b of RG 1.26, explain how the basis of the containment closure action statement would be satisfied following the failure of these components with only the alternate DHR path available.

4. **Boron Dilution:** The piping and instrumentation diagram of the SFP cooling loop indicates that demineralized water can be supplied directly to the refueling water cavity through the SFP cooling system. Evaluation of the temperature distribution within the refueling cavity during operation of the alternate DHR path in WCAP-15872 indicates that cool water entering the refueling cavity from the SFP cooling loop remains stratified, spills over the reactor vessel flange, and flows downward directly into the reactor vessel. Neither the topical report nor the license amendment request explicitly addressed boron dilution from this flow path. Describe how boron dilution from this path affects the boron concentration within the reactor vessel and how continued dilution would be assured of being detected and mitigated prior to dilution below the minimum boron concentration required to maintain an adequate margin to criticality.

5. **Spent Fuel Pool Cooling Reliability:** The SFP pool cooling system is a shared system that cools the two normally connected SFPs at Calvert Cliffs. The system includes two SFP cooling loops operating in parallel, each loop consisting of one half-capacity pump and one half capacity heat exchanger. The alternate DHR path uses one loop to cool the refueling cavity and connected reactor vessel instead of the SFP. However, the Calvert Cliffs Updated Final Safety Analysis Report (UFSAR), Revision 26, states that, in the event one SFP cooling loop is lost, the remaining loop can remove a heat load associated with a recent partial core discharge while maintaining the SFP temperature at 155 EF. The Calvert Cliffs UFSAR also states that a complete loss of cooling is not part of the design basis for the SFP structural components and the SFP cooling system. Explain how these design bases and the requirements of the proposed TS 3.9.4 would be satisfied when one spent fuel cooling loop is operating as an alternate reactor vessel DHR path and no shutdown cooling loop is operable.

6. **Reliability of the Alternate DHR Path:** Describe the reliability of an SFP cooling loop when used as an alternate to an SDC loop. Specifically address vulnerability to blockage of each spent fuel cooling loop suction by rigid (e.g., steel deck material on the refueling bridge) or flexible (e.g., clothing, rubber gloves, and tape) materials that may be located in, over, or around the refueling pool and the potential for significant pump damage as a result of suction blockage. Also, compare the redundancy of flowpaths, pumps, and electrical power supplies of the SFP loop with an SDC loop.

7. **Instrumentation:** GDC 63 specifies that appropriate systems be provided in fuel storage and handling areas to detect conditions that may result in a loss of residual heat removal capability. Describe how the isolation of the SDC loops from the reactor coolant system affect the ability to effectively monitor parameters associated with residual heat removal, such as temperature and pressure (as a surrogate for water level). Describe the frequency and location of monitoring of the refueling cavity and SFP water level when alternate DHR is in operation, and any annunciated alarms associated with these parameters.

Address how this monitoring assures a low probability of a significant inadvertent draindown of the refueling cavity.

9. **Fuel Handling:** The license amendment request indicates that the average temperature of the refueling cavity would be higher with alternate DHR in service than with an SDC loop in service and that the water directly above the reactor vessel would be significantly above the average temperature. Describe how the effect of the resulting higher air temperatures and humidity levels on human performance would be managed to prevent a significant increase in human errors during fuel handling.

10. **TS Bases:** Provide revised bases for the proposed amendment to TS 3.9.4, as specified by 10 CFR 50.90, 10 CFR 50.34(b)(6)(vi), and 10 CFR 50.36(a).

11. **Final Safety Analysis Report:** The NRC staff did not identify any discussion of the alternate DHR path in Final Safety Analysis Report (FSAR) Section 9.2, “Shutdown Cooling System,” or Section 9.4, “Spent Fuel Pool Cooling System.” Identify any FSAR sections that were updated to describe the alternate DHR path following issuance of the associated license amendment on June 16, 1981.

cc:

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