August 26, 1998

Mr. Charles H. Cruse, Vice President
Nuclear Energy Division
Baltimore Gas and Electric Company
1650 Calvert Cliffs Parkway
Lusby, MD 20657-47027

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR THE REACTOR PRESSURE VESSELS AND CONTROL ELEMENT DRIVE MECHANISMS/ELECTRICAL (TAC NOS. M99587, M99588, AND M99206)

Dear Mr. Cruse:

By letter dated July 30, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Reactor Pressure Vessels and Control Element Drive Mechanisms/Electrical System (4.2) integrated plant assessment technical report as attached to the "Request for Review and Approval of System and Commodity Reports for License Renewal." BGE requested that the Nuclear Regulatory Commission staff review the Reactor Pressure Vessels and Control Element Drive Mechanisms/Electrical System (4.2) integrated plant assessment technical report to determine if the report meets the requirements of 10 CFR 54.21(a), "Contents of application-technical information," and the demonstration required by 10 CFR 54.29(a)(1), "Standards for issuance of a renewed license," to support an application for license renewal if BGE applied in the future. By letter dated April 8, 1998, BGE formally submitted its license renewal application.

The NRC staff has reviewed the Reactor Pressure Vessels and Control Element Drive Mechanisms/Electrical System (4.2) integrated plant assessment technical report against the requirements of 10 CFR 54.21(a)(1), 10 CFR 54.21(a)(3). By letter dated April 4, 1996, the staff approved BGE's methodology for meeting the requirements of 10 CFR 54.21(a)(2). Based on a review of the information submitted, the staff has identified in the enclosure, areas where additional information is needed to complete its review.

Please provide a schedule by letter or telephonically for the submittal of your responses within 30 days of the receipt of this letter. Additionally, the staff would be willing to meet with BGE prior to the submittal of the responses to provide clarifications of the staff's requests for additional information.

Sincerely,
Original Signed By

David L. Solorio, Project Manager
License Renewal Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318
Enclosure: Request for Additional Information
cc w/encl: See next page
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REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT
UNIT NOS. 1 & 2
REACTOR PRESSURE VESSELS AND CONTROL ELEMENT DRIVE MECHANISMS/ELECTRICAL INTEGRATED PLANT ASSESSMENT, SECTION 4.2
DOCKET NOS. 50-317 AND 50-318

Section 4.2.1 - Scoping

1. We noted that page 4.3-5 of Section 4.3 indicated that the reactor vessel head lifting rig is discussed with the Fuel Handling Equipment and Other Heavy Load Handling Cranes of Section 3.2 of the license renewal application (LRA). However, Figure 4-2 (Rev.18) provided in Chapter 4 of the Calvert Cliffs Nuclear Power Plant (CCNPP) Updated Final Safety Analysis Report (UFSAR) for Units 1 and 2 shows a component attached to the closure head of reactor pressure vessel, which is called a lifting lug. Are lifting lugs included within the scope of license renewal? If so, provide a cross reference to where they are addressed in the LRA. If not, provide the basis for their exclusion.

2. Figure 4-2 (Rev.18) of the CCNPP UFSAR shows that the closure head insulation is attached to the closure head of reactor pressure vessel. Please describe the functions of closure head insulation, and indicate if the closure head insulation is required to support one of the functions listed in 10 CFR 54.4(a)(1)(i)-(iii).

3. Please clarify whether the component identified in comment (d) of Table 4.2-2 of Section 4.2.1 as a "Core Stop Lug" is same component labeled as the core support lug in Figure 4-2 (Rev.18) provided in Chapter 4 of the CCNPP UFSAR. If these components are not the same, please describe the functions of core support lug and indicate if the core support lug is required to support one of the functions listed in 10 CFR 54.4(a)(1)(i)-(iii).

4. What changes to the scope or other aspects of the Boric Acid Inspection Program have been made in response to the experiences documented in Section 4.2.2 (pg 4.2-14) of the LRA?

Section 4.2.2 - Aging Management

5. Pursuant to 10 CFR Part 50, Appendix G, provide an analysis of the vessel beltline material to demonstrate that they will maintain at least 50 ft-lb Charpy upper-shelf energy (USE) during the period of extended operation, based on the projected neutron fluence and the chemistry of the beltline material. Provide all Charpy USE material data for each beltline material.

6. Provide an outline of the Reactor Vessel Material Surveillance Program and discuss how they will be used to monitor neutron irradiation for the reactor pressure vessel (RPV) beltline materials during the period of extended operation. Provide a summary of "CCNPP Comprehensive Reactor Vessel Surveillance Program (CRVSP)" so that the staff can determine that CRVSP is complete and adequate. Are there supplemental or standby capsules available to be used?

Enclosure

7. How is your assessment of pressurized thermal shock (PTS) affected by the results from the McGuire 1 material surveillance program? Include in your evaluation, the results from the McGuire 1 capsule Y, which is contained in Duke Energy letter to the NRC, dated April 22, 1998.

8. Provide pressure-temperature (P-T) limits for the extended operating term and identify the operating window relative to pump operation for the shutdown cooling system. During the extended licensed term, will there be any limitations in operation of the shutdown cooling system due to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Appendix G, P-T operating limits and the minimum permissible temperature of the reactor vessel?

9. As identified in Section 4.2.1.1 of the submittal under "Unintentional inclusion of Slag Stringer in RPV", the fabrication flaw in the Unit 1 reactor vessel weld was stated to be acceptable in accordance with the applicable ASME Code, Section XI during the preservice and subsequent inservice inspections. However, the flaw acceptance criteria of the Code have been based on a 40-year operating life, equivalent to four inspection intervals of 10-year duration each. Therefore, the flaw should be evaluated analytically for the extended term of operation. Provide an evaluation in accordance with IWB-3600 of the ASME Code, Section XI. Identify the location of the flaw within the weld. If the location of the flaw designates it as a surface planar flaw in the inside surface of the reactor vessel in accordance with the ASME Code, Section XI, paragraph IWA-3310, provide an analysis that demonstrates that PTS including small-break-loss-of coolant accident with an extended high pressure injection transient is not a concern consistent with the bases for 10 CFR 50.61. Provide initial and adjusted reference nil-ductility transition temperature (RT-NDT), delta RT-NDT, margin, neutron fluence, and chemistry (Copper and Nickel) of the weld containing the flaw in accordance with Regulatory Guide 1.99, Rev. 2.

10. Provide a description of the "CCNPP Alloy 600 Program" which is implemented as an aging management program for discovery and mitigation of age related degradation, particularly primary water stress corrosion cracking (PWSCC) in Inconel, and explain how the program will be implemented during the license renewal term.

11. How will BGE determine the condition of partial penetration welds in the vessel head penetrations and in the bimetallic welds of control element drive mechanisms (CEDM) penetration nozzles? In particular, discuss how BGE intends to extend its commitments to Generic Letter 97-01, "Degradation of Control Rod (Element) Drive Mechanism Nozzles & Other Vessel Closure Head Penetrations", over the proposed extended term of operation for the CCNPP units. Include in the discussion an assessment and reanalysis of the CCNPP CEDM nozzles using the latest crack initiation and growth model that was developed for the Combustion Engineering (CE) Owners Group to assess postulated flaws in CE designed CEDM penetration nozzles. With respect to this reanalysis, provide what the probability will be for cracks to have initiated in the CEDM penetration nozzles at the end of the current license and at the end of the proposed extended terms, and state what the anticipated degree of crack growth is for postulated flaws in the CEDM penetration nozzles at the end of the current license and at the end of

the proposed extended operating terms. Identify any volumetric examination of Calvert Cliffs or of other plants CEDM penetration nozzles that will confirm your susceptibility analysis that cracking will not occur during the license renewal term.

12. Table 4.2-2 of the submittal identifies stress corrosion cracking of the RPV flow skirt, as being a plausible age-related degradation mechanism. Discuss how BGE intends to monitor the flow skirt-to-vessel weld for PWSCC during the extended term of operation. To what extent will the flow induced vibration in the flow skirt affect integrity of the subject weld?

13. In Section 4.2.2 of the submittal, you have identified the aging related degradation mechanisms (ARDMs) for various RPV components. Based on these aging mechanisms, how will your inservice inspection (ISI) program be tailored to monitor age related degradation due to these mechanisms for these components? Is there any weld on these components that is not examined due to physical constraints or geometry? Provide your plan to request any relief from the Code-required examination of such welds during the renewal term.

14. Based on its evaluation of operating experience, the NRC has determined that potential aging effect mechanisms in components of PWR vessels are as indicated in the Table 3.1-3 of the Draft Standard Review Plan for License Renewal. Table 3.1-3 identifies components that are considered part of the reactor pressure vessel (RPV) and identifies the associated aging effects for the components. Identify the equivalent components in the Calvert Cliffs RPV and identify the aging effects applicable to these components. Explain how the aging effects that are identified as "Significant" or "Unresolved" in the table are addressed for both Calvert Cliffs RPVs.

15. Section 4.2.2 includes a discussion that the ISI walkdown inspections (VT-2) after reactor shutdown and prior to plant startup must ensure that all components that are the subject of Issue Reports, where boric acid leakage has been found, are examined in accordance with the requirements of the program. Does the scope of components covered by the Boric Acid Inspection Program include all of the components for which general corrosion caused by boric acid is plausible, or only those which have been the subject of Issue Reports?

16. Section 4.3 of the LRA entitled "Reactor Vessel Internals (RVI) System" indicates that the core support barrel snubber and snubber bolts are addressed in this Section 4.2. The NRC staff did not find these devices described in Section 4.2, therefore, please describe how and where these components are addressed in the LRA.

17. Section 4.2.2 of the LRA states "The threshold for onset of neutron effects for RPV materials is conservatively defined to be a fast neutron fluence that exceeds 1E17n/cm^2." citing Appendix H of 10 CFR Part 50. The staff believes that Appendix H cites the indicated neutron fluence as a threshold below which a reactor vessel material surveillance program is not required for the vessel. Appendix H thereby creates in effect a "regulatory threshold" for neutron fluence, but clearly not a mechanistic threshold below

which neutron effects do not occur. Please provide your basis for concluding that there are negligible effects from neutron fluence below 1E17n/cm^2.

18. Inconel alloy and stainless steel components become susceptible to IASCC at neutron fluence greater than 5E20 n/cm^2 . Since the flow skirt or flow baffle is located between the core and the reactor pressure vessel, the component would be expected to experience a large neutron fluence. What is the peak fluence for this component and what are the consequences of neutron embrittlement for this component given any potential susceptibility to irradiation assisted stress corrosion cracking (IASCC) or stress corrosion cracking (SCC)? Are there any other Inconel alloy components (such as the surveillance capsule holders) that receive a sufficiently large neutron fluence (greater than or equal to 5E20 n/cm^2) that are potentially susceptible to IASCC? In such cases, what is the peak fluence for these components and what are the consequences of neutron embrittlement on these components given their potential susceptibility to IASCC (or SCC)?

19. For the components identified with a plausible ARDM, identify any components which are not routinely inspected as a part of the ISI Program or any other program.

20. Section 4.2 indicates that the locations of interest for low cycle fatigue are the RPV main coolant outlet nozzles and closure head flange studs. The report further indicates that all other RPV components and/or subcomponents are considered to have low susceptibility to low-cycle fatigue. Describe the specific criteria used to determine that the other RPV components and/or subcomponents have a low susceptibility to low-cycle fatigue.

21. Section 4.2 indicates that the Fatigue Monitoring Program (FMP) monitors and tracks low-cycle fatigue usage for the selected components of the nuclear steam supply system and the steam generators. Describe the parameters that are monitored by the FMP that are applicable to the RPV. Also describe how the monitored parameters are compared to the fatigue analysis of record.

22. Section 4.2 indicates that in order to stay within the design basis, corrective action is initiated well in advance of the cumulative usage factor approaching one or the number of cycles approaching design allowable. Describe the criteria used to determine when corrective actions will be initiated.

23. Section 4.2 indicates that the FMP "will perform an engineering evaluation to determine if the low-cycle fatigue usage for the Control Element Drive Mechanisms (CEDMs)/Reactor Vessel Level Monitoring System (RVLMS) components are bounded by the existing bounding components." Describe the fatigue criteria used for the design of the CEDM/RVLMS components. Please indicate the reason the FMP is performing an engineering evaluation on these components.

24. Section 4.2 indicates that the current usage factors for the critical RPV components are well below one. Provide the usage factors projected for the critical RPV components at the end of the proposed extended period of operation including a summary discussion of how they were derived.

25. Section 4.2 of the license renewal application indicates that the licensee in conjunction with the Electric Power Research Institute has initiated an additional study to evaluate the effects of low-cycle fatigue on various fatigue critical plant locations. Provide a description of this study and describe its applicability to the Calvert Cliffs RPV and CEDM/RVLMS components.

26. Are there any parts of the systems, structures and components within the RPV or CEDM system that are inaccessible for inspection? If so, describe what aging management program will be relied upon to maintain the integrity of the inaccessible areas. If the aging management program for the inaccessible areas is an evaluation of the acceptability of inaccessible areas based on conditions found in surrounding accessible areas, please provide information to show that conditions would exist in accessible areas that would indicate the presence of, or result in degradation to, such inaccessible areas. If different aging effects or aging management techniques are needed for the inaccessible areas, please provide a summary to address the following elements for the inaccessible areas: (a) Preventive actions that will mitigate or prevent aging degradation; (b) Parameters monitored or inspected relative to degradation of specific structure and component intended functions; (c) Detection of aging effects before loss of structure and component intended functions; (d) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (e) Acceptance criteria to ensure structure and component intended functions; and (f) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.