

50-317/318



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 13, 1998

Mr. Charles H. Cruse, Vice President
Nuclear Energy Division
Baltimore Gas and Electric Company
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CLARIFICATION OF SEVERAL NRC REQUESTS FOR ADDITIONAL
INFORMATION ON CALVERT CLIFFS NUCLEAR POWER PLANT LICENSE
RENEWAL APPLICATION SUBMITTED BY THE BALTIMORE GAS AND
ELECTRIC COMPANY (TAC NOS. MA1445, MA1446, M99214, M99587,
M99588, M99206, M98835, M98837, AND M99181)

Dear Mr. Cruse:

On September 22, 1998, the Nuclear Regulatory Commission (NRC) staff held a public meeting with representatives of Baltimore Gas and Electric Company (BGE) at Rockville, Maryland, to discuss the progress of the NRC staff's review of BGE's License Renewal Application for its Calvert Cliffs Nuclear Power Plants, Units Nos. 1 and 2. During the meeting, BGE requested clarification on approximately 25 requests for additional information (RAI) that were subsequently expanded to 26. BGE requested clarification in order to meet the November 21, 1998, milestone for submitting its responses to all the staff's RAI.

The purpose of this letter is to clarify selected RAI for BGE. The RAI have been revised based on the result of BGE's comments, as provided to the NRC in Enclosure 3 of the September 28, 1998 NRC and BGE meeting summary (which is provided as Enclosure 1 to this letter), and on subsequent discussions between BGE and NRC staff which were held to gain additional detail regarding BGE's needs for clarification. Specifically, RAI numbers 5.10.6 and 4.2.8 are being clarified by this letter.

By telecon on November 5, 1998, the NRC provided BGE with a revised version of RAI 5.10.6. BGE subsequently communicated to the NRC that the NRC had sufficiently clarified the RAI such that BGE could respond. The revised RAI is provided in Enclosure 2.

By telecon on October 30, 1998, the NRC provided BGE a revised version of RAI 4.2.8. By telecon on November 5, 1998, BGE indicated that additional time would be needed for BGE to evaluate its ability to respond to the revised RAI. As a result, BGE committed to provide an interim response, if additional time was needed in developing its response, by the November 21, 1998 milestone. The revised RAI is provided in Enclosure 2.

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November 13, 1998

By facsimile the NRC provided BGE with copies of Enclosure 3 to provide additional clarification regarding RAI 4.3.18. By telecon on November 5, 1998, BGE indicated to the NRC that the additional information did not provide sufficient justification for BGE to change its revised proposed response as provided in Enclosure 1.

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

- Enclosures: 1. Revised Copy of USNRC Requests for Additional Information Provided to NRC by BGE
- 2. Revised Requests for Additional Information
- 3. Corrosion Data for Stainless Steel

cc w/ enclosures: See next page

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**BGE REQUESTS FOR CLARIFICATION
ON
USNRC REQUESTS FOR ADDITIONAL INFORMATION**

Prepared by
BGE Calvert Cliffs License Renewal Project

October 1, 1998

**BGE REQUESTS FOR CLARIFICATION
ON
USNRC REQUESTS FOR ADDITIONAL INFORMATION**

Prepared by
BGE Calvert Cliffs License Renewal Project

October 1, 1998

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
FEEDWATER SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.9
DOCKET NOS. 50-317 AND 50-318

Aging Management

- 5.9.47 One of the most effective ways of minimizing erosion/corrosion is to control secondary water chemistry, that is, pH and oxygen concentration. Describe whether pH and oxygen concentration are controlled in the feedwater system and if so, specify the parameter ranges.

BGE believes that this question has been answered in the LRA as well as has been discussed at length in meetings. BGE has provided the NRC with copies of the procedures. In addition, there is an effort between NEI (NEI-9706) and NRC on chemistry controls which is ongoing that provides a significant amount of information in this area. BGE requests NRC evaluate the need for requesting this additional information, given the above described exchanges already underway.

- 5.9.54 Page 5.9-20 of the application indicates that the Institute of Nuclear Power Operations (INPO) has performed assessment of the BGE erosion/corrosion program and provided recommendation for enhancements. Please briefly summarize the results of the INPO assessment and outline the INPO recommendations for improvements at the Calvert Cliffs plants.

BGE requests that NRC review the need for INPO reports since they are proprietary and the NRC has access to them already.

**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 4.2.2 - Aging Management

- 4.2.8 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?

BGE is requesting NRC clarify #8 above. Current plant practice is to maintain these curves as required, not necessarily at license termination points. No 10CFR50 requirement exists for such submittals other than to maintain them current and not violate them. The 2nd part of this question is hypothetical and based upon 60 year curves.

BGE has had a TELCON with NRC on this question and is providing response according to NRC clarification.

- 4.2.17 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 $1E17n/cm^2$.

BGE is requesting clarification of #17 above. BGE has had a TELCON with NRC on this question and is answering this question according to NRC clarification.

**REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
SPENT FUEL POOL COOLING SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.18
DOCKET NOS. 50-317 AND 50-318**

Section 5.18.2 - Aging Management

- 5.18.10 Provide a summary description of Calvert Cliffs operating and maintenance experience related to boric acid corrosion of carbon steel components. In particular, characterize the extent to which boric acid corrosion of carbon steel components has changed since the initial implementation of the boric acid corrosion inspection (BACI) program. Also, describe the extent to which carbon steel components in the spent fuel pool cooling system have had to be repaired or replaced because of boric acid corrosion since the implementation of the BACI program.

This question is too broad. Similar questions were withdrawn by NRC and refocused. BGE requests the NRC either conduct a site visit and discuss OE with plant personnel or rephrase question such that it is more focused.

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NOS. 1 & 2
REACTOR COOLANT SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 4.1
DOCKET NOS. 50-317 AND 50-318

- 4.1.9 For the following aging effects and components, summarize the extent to which BGE relies upon the associated programs for aging management, and provide examples of any operating experience that demonstrates the effectiveness of the programs that are relied upon to manage these aging effects:
- a. boric acid corrosion -- Technical Specifications (TS) leakage limits, and ASME Section XI, Subsection IWB, examination categories B-P;
 - b. cracking of large bore piping -- ASME Section XI, Subsection IWB, examination categories B-J and B-F, and flaw evaluation criteria IWB-3000;
 - c. cracking of small bore piping (less than 4 in but greater than 1 in diameter) -- augmented volumetric inservice inspection; and, because some safe ends and welds on small bore piping are of Inconel, information resulting from the assessment of NRC Information Notice (IN) 90-10;
 - d. cracking of bolting -- programs consistent with ASME Section XI, Subsection IWB, examination categories B-G-1 and B-G-2, and NRC Bulletin 82-02;
 - e. pressurizer shell, heads, heater belt forgings -- ASME Section XI, Subsection IWB, examination categories B-B and B-P, and primary water chemistry;
 - f. pressurizer nozzles -- ASME Section XI, Subsection IWB, examination categories B-D, B-E, B-F, and B-P, TS leakage limits, primary water chemistry, augmented inspection of small bore piping; and if Inconel is used, information resulting from IN 90-10;
 - g. integral attachments -- ASME Section XI, Subsection IWB, examination category B-H, and primary water chemistry;
 - h. heater sheaths and end caps -- ASME Section XI, Subsection IWB, examination category B-P, and TS leakage limits;
 - i. loss of preload in bolting -- ASME Section XI, Subsection IWB, examination categories B-G-1, B-G-2, and B-P, response to NRC Bulletin 82-02 and Generic Letter 88-05, and TS leakage limits.

BGE believes this question is too broad and requests NRC clarify its intent. An option for disposition is meetings either at Calvert Cliffs or NRC Offices to review the site documentation that may have addressed these issues.

- 4.1.12 It appears that BGE used ferrite criteria to screen components subject to thermal embrittlement. However, the NRC regards ferrite content as inadequate criterion for screening as stated in NUREG-1557. Therefore, justify using ferrite content as screening criteria.

The use of ferrite criteria to screen components has been a part of an Industry Position since 1994. It has also been submitted as part of NEI/EPRI efforts to resolve generic aging issues. BGE requests NRC explain the "inadequateness" of a position.

4.1.17 Please provide a summary description for the following procedures regarding how their implementation will address the following elements for their related aging management program(s): (a) The scope of structures and components managed by the program; (b) Preventive actions designed to mitigate or prevent aging degradation; (c) Parameters monitored or inspected relative to degradation of specific structure and component intended functions; (d) Detection of aging effects before loss of structure and component intended functions; (e) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (f) Acceptance criteria to ensure structure and component intended functions; and (g) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

- a. Procedure SG-20, "Primary manway cover removal and installation"
- b. Administrative Procedure MN-3-110, "Inservice Inspection of ASME XI Components"
- c. Technical Procedure FASTENER-01, "Torquing and Fastener Applications"
- d. Procedure STP-M-574-1/2, "EC Examination of CCNPP 1/2 Steam Generators"
- e. CASS Evaluation program
- f. Alloy 600 program
- g. STP-0-27-1/2, "RCS Leakage Evaluation"
- h. MN-3-301, "BACI Program"
- i. EN-1-300, "Implementation of Fatigue Monitoring"

This question is too broad. Similar questions were withdrawn by NRC and refocused. BGE requests the NRC either conduct a site visit and discuss details of plant programs with plant personnel or re-phrase question such that it is more focused.

**REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
FIRE PROTECTION SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.10
DOCKET NOS. 50-317 AND 50-318**

Section 5.10.1 - Scoping

- 5.10.6 Summarize the changes to the post-fire safe shutdown analysis and the fire hazards analysis that have been implemented since plant licensing and briefly discuss how the analyses, including changes, were addressed in the system level scoping process.

BGE believes this question is too broad and requests NRC clarify its intent. BGE interprets this question as a compilation of CLB and BGE is not clear on that is what NRC intends nor to its contribution to scoping results.

**REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT UNITS NOS. 1 & 2
SERVICE WATER SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.17
DOCKET NOS. 50-317 AND 50-318**

- 5.17.7 The rate of corrosion of the components in the SRW system can be mitigated by proper control of the water chemistry. Provide the specifications for the water chemistry in the SRW system. Include the target values for the individual parameters and their monitoring frequency.

BGE believes this question is too broad and requests NRC clarify its intent. An option for disposition is meetings either at Calvert Cliffs or NRC Offices to review the site documentation that may have addressed the attributes or details of the program but not discussed them in the LRA since they were referenceable.

**REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
TIME-LIMITED AGING ANALYSES, SECTION 2.1
DOCKET NOS. 50-317 AND 50-318**

- 2.1.3 Page 2.1-4 of the license renewal application (LRA) indicates that the pressure-temperature (P-T) limits in the Calvert Cliffs Technical Specifications are valid for Units 1 and 2 for 48 and 30 effective full power years, respectively. Section 4.2 of Appendix A to the BGE application indicates that the Unit 2 reactor vessel is less susceptible to neutron embrittlement. Discuss why the P-T limits for Unit 2 are valid for a shorter time period than for Unit 1. Also, discuss whether the existing P-T limits "[i]nvolve time-limited assumptions defined by the current operating term, for example, 40 years." (Criterion 3 of the definition of TLAA in 10 CFR 54.3(a))

See #8 in Reactor Vessel/CEDM Section. This is a duplicate.

- 2.1.4 10 CFR 54.21(c) requires an evaluation of TLAAs as part of the contents of an LRA. However, Section 2.1 of Appendix A to the BGE application contains future commitments to perform the TLAA evaluations. The following are examples:

| Subsection | Heading | Statement |
|------------|--|---|
| 2.1.3.2 | Irradiation Embrittlement | "... will continue to be updated..." |
| 2.1.3.5 | Containment Liner Plate Fatigue Analysis | "This review ... will be projected ... by the year 2012." |
| 2.1.3.6 | Containment Tendons Prestress Loss | "... recalculated by the year 2012..." |
| 2.1.3.7 | Poison Sheets in Spent Fuel Pool | "This analysis is currently being updated..." |

In accordance with 10 CFR 54.21(c)(1)(iii), describe how BGE will ensure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

BGE is responding to this question by referring to its commitment management procedure. This has been discussed with NRC Branch Chief.

**REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS UNITS 1 AND 2 INTEGRATED PLANT ASSESSMENT
ON METAL FATIGUE
DOCKET NOS. 50-317/50-318**

Section 5.2, "Chemical and Volume Control System"

7.6 Section 3.2.3 of EPRI Report TR-107515 contains an evaluation of environmental effects on the CVCS Charging Inlet Nozzle using methodology developed in EPRI Report TR-105759, "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations," dated December 1995. The attached evaluation summarizes the staff's technical concerns regarding the methodology in EPRI Report TR-105759. Attached are comments on the application of the EPRI methodology for environmental fatigue factors to the Calvert Cliffs plant. Based on these comments, provide the following:

- (a) Discuss the impact of the current Argonne National Laboratory (ANL) statistical correlations of environmental test data on the Calvert Cliffs fatigue evaluation.

BGE does not believe the subject research project can be commented on in a timely nor reasonable fashion. BGE requests NRC withdraw this question since commenting on excerpts from research activities can produce out of context conclusions.

- (b) The technical basis for the assertion that the American Society of Mechanical Engineers (ASME) Code stainless steel fatigue design curve contains sufficient margin to accommodate moderate environmental effects. Include a discussion of the factor required to adjust the laboratory test data for size and surface finish effects and the margin necessary to account for scatter of the test data.

BGE requests the NRC withdraw this question. BGE accepts the ASME code as endorsed by 10CFR50.55a as part of our CLB.

- (c) The technical justification for the strain threshold values.

BGE will provide this answer.

Section 4.1, "Reactor Coolant System"

7.15 Section 4.1 of the application indicates that environmental effects do not apply to the RCS components because of the low oxygen concentrations and because the RCS carbon steel interior surfaces are clad with stainless steel. Discuss the applicability and impact of the latest stainless steel fatigue correlation from ANL on this conclusion (see attachment).

BGE does not believe the subject research project can be commented on in a timely nor reasonable fashion. BGE requests NRC withdraw this question since commenting on excerpts from research activities can produce out of context conclusions.

- 7.16 Section 3.3.3 of EPRI Report TR-107515 contains an evaluation of the Surge Line using methodology developed in EPRI Report TR-105759. Discuss the applicability and impact of the latest stainless steel fatigue correlation from ANL on this evaluation (see attachment).

BGE does not believe the subject research project can be commented on in a timely nor reasonable fashion. BGE requests NRC withdraw this question since commenting on excerpts from research activities can produce out of context conclusions.

- 7.17 Section 3.3.3.2 of EPRI Report TR-107515 indicates that the procedure in Section 3.1.3.2 of the EPRI report was used to develop the environmental factor used in the evaluation. Indicate whether the factor was calculated based on a "standard" treatment or "weighted average" approach as discussed in a June 1, 1998, letter from the Nuclear Energy Institute to the NRC regarding EPRI Report TR-105759. If the "weighted average" approach was used, provide the test data used to develop the approach. Include a statistical assessment of the test data scatter. Compare the results of the statistical assessment with the ANL assessment contained in NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Ferritic Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments." On the basis of this comparison, indicate whether the use of the "weighted average" approach will produce an adequate margin to account for test data scatter.

BGE does not believe the subject research project can be commented on in a timely nor reasonable fashion. BGE requests NRC withdraw this question since commenting on excerpts from research activities can produce out of context conclusions.

Section 5.15, "Safety Injection System"

- 7.22 Section 5.15 of the application indicates that environmental effects do not apply to the SI components because of the low oxygen concentrations and the stainless steel components materials used in fabrication of the affected piping and valve subcomponents. Discuss the applicability and impact of the latest stainless steel fatigue correlation from ANL on this conclusion (see attachment).

BGE does not believe the subject research project can be commented on in a timely nor reasonable fashion. BGE requests NRC withdraw this question since commenting on excerpts from research activities can produce out of context conclusions.

**REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
INTEGRATED PLANT ASSESSMENT ON GENERIC SAFETY ISSUES
DOCKET NOS. 50-317 AND 50-318**

- 8.3 In a letter dated June 2, 1998, the staff concluded that license renewal applicants can address GSI-168, "Environmental Qualification of Electrical Equipment," by providing a technical rationale demonstrating that the current licensing basis for EQ pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. The NRC staff has not completed guidance on the information necessary to demonstrate adequate aging management for the EQ time limited aging analyses (TLAAs). Until that matter is resolved, please provide the EQ Master List of electrical equipment and indicate which of the TLAA categories in 10 CFR 54.21(c)(1) apply to each of the electrical equipment groups. In addition, summarize the procedures that are used to maintain compliance with the requirements of 10 CFR 50.49, and justify that those procedures will adequately manage the EQ analyses for the period of extended operation.

BGE has provided all the requested information in the LRA section on EQ except for the EQ Master List. BGE requests to discuss this with NRC since compliance with 10CFR50.49 is requirement that will carry forward as part of the CLB in accordance with the rule. The EQ Master List is maintained and available on site. Providing such a list would be redundant, require additional regulatory controls beyond 10CFR50.49 and will unnecessarily burden BGE.

**REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
REACTOR VESSEL INTERNALS
INTEGRATED PLANT ASSESSMENT, SECTION 4.3
DOCKET NOS. 50-317 AND 50-318**

Section 4.3.2 - Aging Management

- 4.3.18 Table 4.3 indicates that many components (CEASB, CS, CSTR, CSB, CSC, CSP, FAPFP, and LSSBA) are susceptible to neutron embrittlement, which generally results in loss of fracture toughness in the material composing the component. This loss of fracture toughness is a reduction in resistance to crack growth, which could mean that parts that are macroscopically degraded (through wear or some sort of cracking mechanism such as SCC or fatigue) may fail (fracture) at load levels and/or degradation (i.e., smaller crack sizes) that are lower than those if the part was not in an embrittled condition. Identify for each component that is susceptible to neutron embrittlement, the peak neutron fluence at the end of the extended period of operation, and the materials used to fabricate the specific component. For the limiting component (considering the neutron fluence, material fracture toughness and operating stresses in determining the limiting component), provide a fracture mechanics analysis to determine the critical flaw size during normal operation and emergency and faulted conditions. Provide data to justify the fracture toughness assumed in the analysis. Identify the inspection procedure and the capability of the inspection to detect flaws smaller in size than that of the critical flaw.

BGE requests clarification from NRC on this question. BGE's LRA already provides for inspections for these aging effects. The proposed analysis appears to assume these aging effects are somehow unique to license renewal. In addition, BGE believes the overall inspections proposed are more conservative than using an analytically bounding location approach.

**REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NOS. 1 & 2
COMPONENT SUPPORTS AND
PIPING SEGMENTS THAT PROVIDE STRUCTURAL SUPPORT
COMMODITY REPORTS, SECTIONS 3.1 AND 3.1A
DOCKET NOS. 50-317 AND 50-318**

Section 3.1.2 - Aging Management Review

3.1.18 Please clarify the following concerns regarding the information described in Table 3.1-3:

- a. The loading due to rotating/reciprocating machinery has the potential to affect many of the supports listed in the table. Provide the basis for the "N/A" and "not plausible" determination for supports other than electrical raceways, electrical cabinets and instruments, and tanks potentially affected by rotating/reciprocating machinery loads.
- b. Provide the basis for the "not plausible" determination for piping frame and stanchion supports and for metal spring isolators and fixed base supports potentially affected by loading due to hydraulic vibration or waterhammer.
- c. Provide the basis for the "not plausible" and "N/A" determination for piping frame and stanchion supports, for metal spring isolators and fixed base supports, and for loss-of-coolant accident restraints potentially affected by loading due to thermal expansion of piping and/or components.
- d. Provide the basis for the "not plausible" determination for supports potentially affected by stress corrosion cracking of high strength bolts.
- e. Provide the basis for the "not plausible" determination for supports potentially affected by radiation embrittlement of steel.
- f. Provide the basis for the "not plausible" determination for supports potentially affected by grout/concrete local deterioration.
- g. Provide the basis for the "not plausible" determination for supports potentially affected by lead anchor creep.

This question is too broad. Similar questions were withdrawn by NRC and refocused. BGE requests the NRC either conduct a public meeting or a site visit and discuss details of aging effect plausibility calls with plant personnel or re-phrase question such that it is more focused.

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
PRIMARY CONTAINMENT STRUCTURE, SECTION 3.3A
TURBINE BUILDING STRUCTURE, SECTION 3.3B
INTAKE STRUCTURE, SECTION 3.3C
MISCELLANEOUS TANK AND VALVE ENCLOSURES, SECTION 3.3D
ELECTRICAL COMMODITIES, 6.2
DOCKET NOS. 50-317 AND 50-318

General Questions Related to Sections 3.3B, 3.3C, 3.3D, 3.3E and 6.2

- 3.3.9 Provide the details of specific national codes and standards (e.g., ACI, AISC, etc.) including their editions that will be used to determine repairs and acceptance criteria. If there are changes with respect to specific national codes and standards previously committed to as part of the initial licensing basis, describe plans for incorporating these changes in the CCNPP Updated Final Safety Analysis Report.

BGE requests the NRC clarify this question. BGE finds it difficult to identify specific codes and standards that would be used in corrective actions for unidentified or hypothetical deficiencies. BGE also finds the request to reconcile changes to the licensing basis that may have involved codes and standards, or changes to these codes incorporate into the CLB difficult to respond to.

**REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NOS. 1 & 2
EMERGENCY DIESEL GENERATOR SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.8
DOCKET NOS. 50-317 AND 50-318**

Section 5.8.2 - Aging Management

- 5.8.7 Discuss the corrosion allowances in the design of EDG system components that are subject to corrosion, and how they will be addressed as part of the aging management program.

BGE is answering this, as well as similar RAIs, but suggests discussions with NRC to clarify any concerns it has. It is not apparent to BGE the significance of corrosion allowances in any of the CCNPP LRA findings.

- 5.8.8 Page 5.8-1 of the report states that operating experience relevant to aging was obtained based on Calvert Cliffs Nuclear Power Plant specific information and past experience. Describe the basis upon which Baltimore Gas and Electric Company concluded that cavitation corrosion, intergranular attack, stress corrosion cracking, and thermal damage were not plausible aging effects for EDG systems in relation to any industry-wide experience with these aging effects in EDG systems.

This question is too broad. Similar questions were withdrawn by NRC and refocused. BGE requests the NRC either conduct a site visit and discuss OE with plant personnel or rephrase question such that it is more focused.

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NOS. 1 & 2
INTEGRATED PLANT ASSESSMENT
DOCKET NOS. 50-317 AND 50-318

Water Chemistry Program

The following questions apply to the secondary water chemistry as discussed in Section 5.12, "Main Steam and Blowdown System," and Section 5.9, "Feedwater System," of Appendix A to the Baltimore Gas and Electric Company (BGE) license renewal application:

- 9.1. Control of the secondary water chemistry plays an important role in ensuring that steam generators and other components exposed to secondary water will not be damaged by corrosion and will preserve their integrity. Please include the following information on your secondary water chemistry control program:
- a) What amine is being used for controlling pH in the secondary water system?
 - b) Specify major differences in the secondary water chemistry (feedwater and/or steam generator) for power operation, startup, and shutdown.
 - c) Describe and provide technical bases for any significant differences in secondary water chemistry parameters specified in the BGE CP-217 procedure and the values recommended by the Electric Power Research Institute (EPRI) in their guideline reports, referenced in Section 5.12 of Appendix A to the BGE license renewal application.
 - d) Specify the upper limits of the major chemistry parameters and the allowable time period to restore chemistry parameters to acceptable limits.

This question is too broad. Similar questions were withdrawn by NRC and refocused. BGE requests the NRC either conduct a site visit and discuss these questions with plant personnel or re-phrase question such that it is more focused.

**REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
INTEGRATED PLANT ASSESSMENT, SECTIONS 4.1, 4.2, 5.2, 5.7, 5.15, AND 5.16
DOCKET NOS. 50-317 AND 50-318**

Section 4.1, "Reactor Coolant System," and Section 4.2, "Reactor Pressure Vessels and CEDMs/Electrical Systems"

- 4.1.26 Provide the results of BGE's most recent internal audit of the Alloy 600 program; including areas of strengths and weaknesses, safety implication of findings, and corrective action plans and schedule for implementation.

BGE requests the NRC clarify this question. It is not common practice to docket licensee internal audits. Rather, these audits are available for NRC inspection on site and are typically summarized in monthly resident inspector reports.

Section 5.2, "Chemical and Volume Control System"

- 5.2.3 Provide the results of BGE's most recent internal audit of the BACI Program; including areas of strengths and weaknesses, safety implication of findings, and corrective action plans and schedule for implementation.

BGE requests the NRC clarify this question. It is not common practice to docket licensee internal audits. Rather, these audits are available for NRC inspection on site and are typically summarized in monthly resident inspector reports.

**Clarification on Staff Requests for Additional Information
Regarding Baltimore Gas and Electric's Application for License Renewal
For the Calvert Cliffs Nuclear Power Plants Units 1 & 2**

The following are revised versions of RAI as provided to BGE. The previous version of the RAI are given in enclosure 1.

- 5.10.6 Verify that the fire protection screening tool has been updated since its inception and that changes to the UFSAR, fire protection program documentation, and the Interactive Cable Analysis (reference Section 3.3.2 of FSE dated April 4, 1996) have been reviewed and captured in the fire protection screening tool as appropriate.
- 4.2.8 It appears that the current Unit 1 Pressure temperature (P-T) limits are applicable for 60 years. Confirm that the current Unit 1 P-T limits are applicable for 60 years. If the current Unit 1 P-T limits are not applicable for 60 years, will the Unit 1 have sufficient operating window (difference between P-T limits and reactor coolant pump seal limits) to operate the plant at the end of the license renewal term.

FRACTURE TOUGHNESS DATA FOR IRRADIATED STAINLESS STEEL

The fracture toughness of austenitic stainless steel can become degraded with high levels of neutron irradiation, for example fluences greater than 1×10^{20} n/cm² (E > 1MeV). Fracture toughness data for irradiated stainless steels at such high fluences is not plentiful. Two sources available in the public literature are references 1 and 2.

The data in Reference 1 are for the initiation fracture toughness (i.e., at the initiation of crack growth), defined by:

$$K_{Jc} = \sqrt{J_{Jc} \times E}$$

J_{Jc} is defined as the J-integral value at the initiation of crack growth and E is the Young's modulus for the material. For Type 304 stainless steel plate irradiated to a fluence of $\sim 5 \times 10^{20}$ n/cm² (E > 1MeV) at $\sim 280^\circ\text{C}$ and tested at 288°C , the lowest reported value of J_{Jc} (~ 75 in.-lb/in.²) corresponds to a K_{Jc} of ~ 50 ksi $\sqrt{\text{in}}$.

From Reference 2, J-integral resistance or J-R curve data are reported for two samples fabricated from core shroud material removed from an overseas BWR (see the attached figure). The fluence for these samples is reported in Ref. 2 as 8×10^{20} n/cm².

Reconciliation of the J_{Jc} from Reference 1 with the J-R curve trends from Reference 2 (through scaling of the J levels in the J-R curves) can provide one estimate of the fracture toughness of highly irradiated austenitic stainless steel.

Provide any other fracture toughness data used in this evaluation.

REFERENCES

1. Loss, F. J., and Gray, Jr., R. A., "J-Integral Characterization of Irradiated Stainless Steels," NRL Report 7565, Naval Research Laboratory, Washington, D. C., April 25, 1973.
2. EPRI TR-107079, "BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guideline, Revision 2 (BWRVIP-01)," October 1996, pp. 4-13.

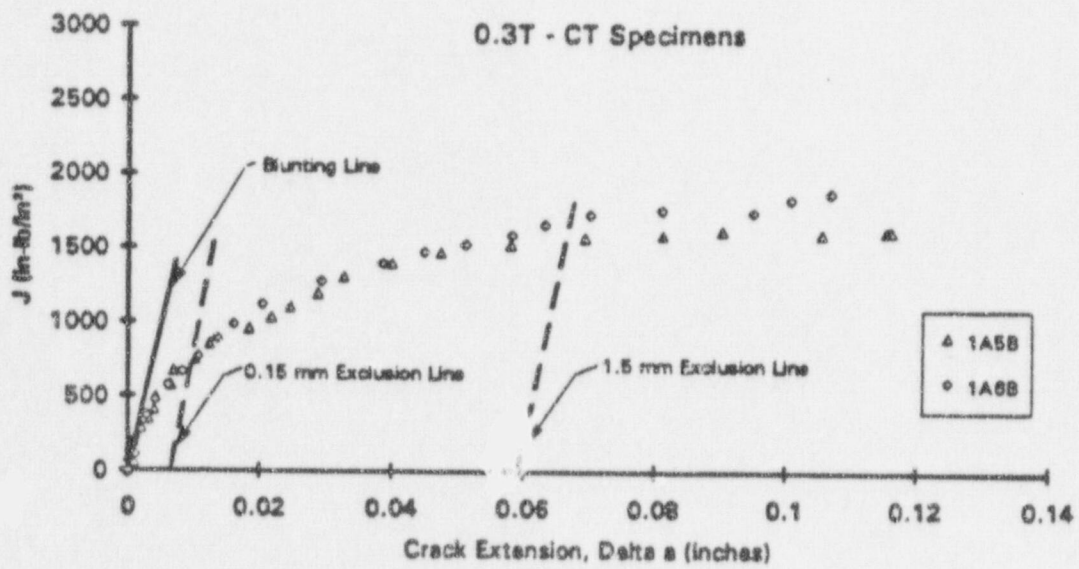


Figure J-R curve data for irradiated stainless steel for a fluence of $8 \times 10^{20} \text{ n/cm}^2$ (Ref. 2).