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October 7, 1998

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	**Docket Nos. 50-317-LR**
	)	**50-318-LR**
BALTIMORE GAS	)	**(License Renewal)**
& ELECTRIC CO., et al.,	)	
	)	**ASLBP No. 98-749-01-LR**
(Calvert Cliffs Unit 1 and	)	
Unit 2)	)	

PETITIONER'S NOTICE OF FILING

Petitioner, the National Whistleblower Center (NWC), by and through counsel, hereby notifies the Licensing Board of significant information that impacts this proceeding. Additionally, petitioner believes the attached information provides a basis for the Board's dismissal of the Baltimore Gas & Electric Company's (BGE) license renewal application to operate Calvert Cliffs Nuclear Power Plant (CNPP) Unit 1 and Unit 2, or in the alternative, for the Board's vacating and rescheduling of the pre-hearing conference that is scheduled to take place on November 12, 1998. *See*, Licensing Board Memorandum and Order (Sep. 29, 1998).

On October 1, 1998, petitioner notified the Board that on September 25, 1998 NWC learned for the first time that the NRC staff had requested BGE to provide a significant amount of information concerning BGE's license renewal application. *See*, Exhibit 1, Letter and Request for Additional Information from NRC staff to BGE (Aug. 28, 1998), attached to Petitioner's Motion to Vacate Pre-Hearing Conference. Notably, this letter was not publicly available in the NRC Public Document Room (PDR) until September 22, 1998. *Id.* Nor did either the NRC staff or BGE notify the Board or the parties about the NRC's RAI.

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On October 7, 1998, NWC learned for the first time about eighteen (18) additional RAI's sent by the NRC staff to BGE about the pending license renewal application. *See*, Exhibits 1-18, attached hereto. The exhibits consist of the following documents:

Exhibit 1: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 31, 1998) (w/attachment);

Exhibit 2: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 1, 1998) (w/attachment);

Exhibit 3: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 1, 1998) (w/attachment);

Exhibit 4: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 1, 1998) (w/attachment);

Exhibit 5: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 2, 1998) (w/attachment);

Exhibit 6: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 2, 1998) (w/attachment);

Exhibit 7: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 2, 1998) (w/attachment);

Exhibit 8: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 3, 1998) (w/attachment);

Exhibit 9: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 3, 1998) (w/attachment);

Exhibit 10: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 3, 1998) (w/attachment);

Exhibit 11: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 4, 1998) (w/attachment);

Exhibit 12: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 4, 1998) (w/attachment);

Exhibit 13: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 4, 1998) (w/attachment);

Exhibit 14: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 7, 1998) (w/attachment);

Exhibit 15: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 7, 1998) (w/attachment);

Exhibit 16: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 7, 1998) (w/attachment);

Exhibit 17: Craig to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 9, 1998) (w/attachment);

Exhibit 18: Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 2, 1998) (w/attachment).

Most all of these additional 18 RAI's were received by the PDR on or after October 1, 1998 -- the deadline imposed by this Board for NWC to submit its amended/supplemental petitioner to intervene and its list of contentions concerning BGE's license renewal application. There are now 19 RAI's known to exist, including the RAI referenced in petitioner's Motion to

Vacate the Pre-Hearing Conference (Oct. 1, 1998) and the additional 18 RAI's discovered by NWC today.

A review these additional 18 RAI's demonstrate that most of BGE's license renewal application requires clarification and supplementation and the NRC staff has raised additional generic questions about the entire renewal application. *Id.* A comparison of these 18 additional RAI's with the BGE renewal application reveals that the NRC has raised serious questions about more than 70 per cent of the technical sections of the renewal application.[1]

Although the 19 RAI's were sent by the NRC staff to BGE between August 28, 1998 and September 9, 1998 none of them were immediately publicly available in the NRC PDR, and most were not even received by the PDR until on or after October 1, 1998. More significantly, neither the NRC staff nor BGE have made this Licensing Board aware of the existence of these 19 RAI's. In addition, the NRC staff failed to include NWC on the service lists attached to any of the 19 RAI's.[2]

It is unknown to NWC how many untold other RAI's may exist concerning the BGE renewal application. It is also unknown when BGE will ever submit its responses to these 18

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[1]These additional 18 RAI's specifically reference 14 of the 19 Sections contained in Volume 1, and 14 of 20 Sections contained in Volume 2, of the BGE renewal application. Moreover, the NRC staff has raised generic questions impacting the entire BGE renewal application. All of the questions raised by the 18 additional RAI's will unquestionably impact BGE's environmental report contained in Volume 3 of the renewal application, because the technical issues raised by the RAI's will affect any evaluation of the increased risk of accidents and other issues involved in the Environmental Impact Statement process.

[2]Even more disturbing is that NWC contacted counsel for BGE on September 25, 1998, to specifically inquire about the one RAI that NWC had discovered on that date (*see,* Ex. 1 to NWC's Oct. 1, 1998 Motion to Vacate), and counsel for BGE did not reveal the existence of any other RAI's. Surely on September 25, 1998 BGE and its counsel were aware there were other RAI's sent by NRC staff.

additional RAI's; however, it seems unlikely that BGE will submit responses to these highly technical questions concerning more than 70% of its renewal application prior to the currently scheduled November 12, 1998 pre-hearing conference.

To make matters worse, NWC just discovered today that on September 15, 1998, David L. Solorio, the NRC staff's Project Manager for license renewal, sent a memorandum to Christopher I. Grimes, Director of NRC License Renewal Project Directorate, about a "forthcoming meeting" to be held on September 28, 1998 between the NRC staff and BGE on the pending license renewal application. *See*, Exhibit 19, attached hereto. Although Mr. Solorio attached a lengthy service sheet to his September 15, 1998 memorandum, the service list did not include NWC or its representatives. Notably, on page 1 of the Mr. Solorio's memo he states:

> Meetings between NRC technical staff and applicants or licensees are open for interested members of the public, intervenors, or other parties to attend as observers pursuant to "Commission Policy Statement on Staff Meetings Open to the Public" 59 *Federal Register* 48340, 9/20/94.

Ex. 19, p. 1 (emphasis added).

At the time that Mr. Solorio sent the above-referenced memorandum NWC was known by the NRC staff to be an "intervenor" in the above-captioned proceeding. Nonetheless, the NRC staff failed to notify the NWC or its representatives about this September 28, 1998 meeting with BGE about the renewal application.

The failure on the part of either the NRC staff or BGE to inform this Licensing Board about the 19 RAI's and the serious deficiencies with BGE's renewal application referenced therein, is highly prejudicial to NWC. BGE's and NRC staff's previous arguments that NWC should have submitted its list of contentions on September 11, 1998 make a mockery of the public

participation provisions of the Atomic Energy Act and NRC regulations because both BGE and NRC staff knew of the existence of the numerous RAI's that had been sent to BGE about the renewal application. It is unfair to NWC to require it to expend time and resources evaluating BGE's renewal application for the purpose of developing contentions when the NRC staff's RAI's raising questions about more than 70% of the renewal application and other generic issues related thereto were not even communicated to NWC.

The Commission's regulations presume that at the time of publication of the required Notice of Hearing that the renewal application is "complete and acceptable" for docketing. *See*, 10 C.F.R. §§ 2.100, 2.101, 2.104. Unquestionably, based on the 19 known RAI's BGE has simply not filed necessary technical information to support its renewal application and it is, therefore, not "complete and acceptable" pursuant to NRC regulations.

Moreover, NRC regulations envision a licensee providing additional information in support of a renewal application within 30 days from the date of request by the NRC staff, and provide for denial of the application if the licensee fails to provide the information within that time. 10 C.F.R. § 2.108(c). On the basis of what is known to NWC, BGE has not responded to the 19 RAI's within 30 days of those requests by NRC staff and it is unknown what alternative scheduling arrangements, if any, may have been agreed to by the NRC staff and BGE concerning BGE's RAI responses. It is obvious that BGE's failure to file a "complete and acceptable" renewal application and the deficiencies contained in application causing the NRC staff to issue at least 19 RAI's violates the Commission's guidelines for expediting renewal licensing proceedings. Although NWC does not accept the validity of those Commission guidelines, if they are to be

equitably applied to the parties in this proceeding the Licensing Board should deny the renewal application due to BGE's failure to meet those standards.

In the event that BGE's renewal application is not denied, it is obvious that the pre-hearing conference cannot proceed as scheduled. NWC should not be required to submit its list of contentions or its supplemental/amended petition until at least 100 days after BGE provides it responses to the RAI's.

Respectfully submitted,

Stephen M. Kohn
3233 P Street, N.W.
Washington, D.C. 20007
(202) 342-2177

Attorneys for Petitioner National Whistleblower Center

October 7, 1998

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CERTIFICATE OF SERVICE

I hereby certify that the foregoing document was served this October 7, 1998 on the following persons by fax (without attached exhibits) and mailed to the parties on October 8, 1998.

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Marian Zobler
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Office of Commission Appellate Adjudication
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Stephen M. Kohn

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50-317
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August 31, 1998

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR THE MAIN STEAM, STEAM GENERATOR BLOWDOWN, EXTRACTION STEAM, AND NITROGEN AND HYDROGEN SYSTEMS (TAC NOS. MA0297, MA0304, AND M99213)

Dear Mr. Cruse:

By letter dated October 22, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Main Steam, Steam Generator Blowdown, Extraction Steam, and Nitrogen and Hydrogen Systems (5.12) 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 (NRC) staff review report 5.12 to determine if the report meet 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 staff has reviewed report 5.12 against the requirements of 10 CFR 54.21(a)(1) and 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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Exhibit 1

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Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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EHackett (EMH1)
AMurphy (AJM1)
TMartin (TOM2)
DMartin (DAM3)
GMeyer (GWM)
WMcDowell (WDM)
SStewart (JSS1)
THiltz (TGH)
SDroggitis (SCD)
DSolorio (DLS2)
PDLR Staff

TMarsh/GHubbard (LBM/GTH)
WLefave (WTL1)
GGeorgiev (GBG)
TSullivan (EJS)
KParczewski(KIP)
RWessman (RHW)
Yeuh-Li Li (YCL)
SLittle (SSL)

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Baltimore Gas & Electric Company
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50-317

September 1, 1998

Mr. Charles H. Cruse, Vice President
Nuclear Energy Division
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Lusby, MD 20657-47027

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'98 OCT -1 P3:44

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT REPORTS FOR THE AUXILIARY FEEDWATER SYSTEM (TAC NOS. MA0295, MA0296, AND M99215)

Dear Mr. Cruse:

By letter dated October 22, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Auxiliary Feedwater System (5.1) 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 (NRC) staff review the Auxiliary Feedwater System (5.1) 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 Auxiliary Feedwater System (5.1) integrated plant assessment technical report against the requirements of 10 CFR 54.21(a)(1) and 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). By letter dated August 21, 1998, the NRC forwarded requests for additional information on scoping to BGE in order to give BGE additional time to prepare its responses while the staff was continuing its review of the subject report. Based on the continuted review of Section 5.1, the staff has identified in the enclosure, areas where additional information related to aging management is needed to complete its review.

Please provide a schedule by letter or telephonically for the submittal of your responses within 30 days of 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.

Exhibit 2

Sincerely,

Original Signed By

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

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Docket Nos. 50-317 and 50-318
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 1, 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, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT REPORTS FOR THE AUXILIARY FEEDWATER SYSTEM (TAC NOS. MA0295, MA0296, AND M99215)

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Sincerely,

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

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NOS. 1 & 2
AUXILIARY FEEDWATER SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.1
DOCKET NOS. 50-317 AND 50-318

Section 5.1.2 - Aging Management

1. The potential and plausible age related degradation mechanisms for the Auxiliary Feedwater (AFW) system are identified in Table 5.1-2 of Section 5.1 of the license renewal application. However, components such as the AFW piping, pumps and valves are considered to have low susceptibility to fatigue. Provide a description of the evaluation and any specific criteria from which you concluded that fatigue is not a plausible aging effect for the AFW components. Inasmuch as corrosion and pitting have been identified as plausible aging effects for the AFW components, include in your response a discussion that effect of the degradation caused by corrosion and pitting on the structural integrity of the components and the basis for excluding fatigue as a plausible aging effect.

2. The pumps and piping in the AFW were judged to have low susceptibility to dynamic loadings. However, based on operating experience, it is likely that the AFW system will be subject to dynamic loads during transient operation and abnormal events such as water hammer. Provide a summary of the evaluations from which you concluded that damage due to dynamic loading is not an aging concern for the critical components in the AFW system during the proposed period of extended operation.

3. Provide a description of the evaluation and any specific criteria from which you concluded that erosion/corrosion is not a plausible aging effect for the components of the AFW system.

4. Identify differences between the Diesel Fuel Oil system and AFW system buried pipe inspection programs. If there are any differences, provide a description and justification for each of the differences.

5. Are there any parts of the systems, structures and components 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.

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RCorreia (RPC)
RLatta (RML1)
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GMeyer (GWM)
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SStewart (JSS1)
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September 1, 1998

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1650 Calvert Cliffs Parkway
Lusby, MD 20657-47027

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT FOR THE FEEDWATER SYSTEM (TAC NOS. M95453, M95454, AND M99178)

Dear Mr. Cruse:

By letter dated May 23, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Feedwater System (5.9) 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 (NRC) staff review the Feedwater System (5.9) integrated plant assessment technical report to determine if the report meet 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 reviewed the Feedwater System (Section 5.9) integrated plant assessment against the requirements of 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(3). By letter dated February 13, 1998, the staff forwarded requests for additional information to BGE in order to give BGE additional time to prepare its responses while the staff was continuing its review of the subject report. Based on the continued review of Section 5.9 of BGE's license renewal application, the staff has identified in the enclosure additional areas beyond those outlined in the February 13, 1998, letter where 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

Enclosure: As stated
Docket Nos. 50-317, 50-318
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Exhibit 3

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Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
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LDoerflein (LTD)
BBores (RJB)
SDroggitis (SCD)
RArchitzel (REA)
CCraig (CMC1)
LSpessard (RLS)
RCorreia (RPC)
RLatta (RML1)
EHackett (EMH1)
AMurphy (AJM1)
TMartin (TOM2)
DMartin (DAM3)
GMeyer (GWM)
WMcDowell (WDM)
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PDLR Staff

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KParczewski (KIP)
SLittle (SLL)

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS UNITS 1 AND 2 FEEDWATER SYSTEM INTEGRATED PLANT ASSESSMENT, SECTION 5.9
DOCKET NOS. 50-317/50-318

Aging Management

1. The BGE application indicates that thermal stratification is a significant contributor to fatigue usage for the steam generator nozzle and adjacent piping. The application further indicates that the piping adjacent to the Unit 2 steam generator was instrumented with thermocouples to obtain temperature data around the circumference of the pipe. Provide a sketch of the piping showing the locations of the thermocouples used to measure the temperature data for the steam generator nozzle.

2. The application indicates that the effect of local thermal stratification in the feedwater system does not extend beyond the first elbow of the vertical pipe run. Provide the basis for this conclusion.

3. The application indicates that a finite element analysis of the steam generator nozzle region was performed to determine the most critical location for fatigue. Provide the following information regarding the finite element analysis:

   (a) Indicate the computer code used for the analysis. Describe the method used to verify the computer code.

   (b) Provide a description of the model used for the analysis and indicate the assumptions used in the analysis. Include a discussion of stress intensification factors, if any, used in the analysis.

4. The application indicates that the critical feedwater nozzle welds in Unit 1 were inspected in 1996, and that no flaws with sizes above the critical flaw size specified in the ASME Code were identified. Characterize the indications, if any, that were identified during the inspections. The application also indicates that Unit 2 welds were scheduled for inspection during the 1997 refueling outage. Provide the results of the Unit 2 inspections including a characterization of any indications identified during the inspections.

5. Electric Power Research Institute (EPRI) Report TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant," provides the results of the fatigue analyses of the feedwater nozzles. Table 3-16 of the EPRI report indicates that fatigue usage factors, without considering environmental effects, will exceed 1.0 prior to forty years of operation for two Unit 2 steam generator nozzles . Section 3.1.4 of the EPRI report contains a flaw tolerance evaluation in accordance with criteria in ASME Section XI Nonmandatory Appendix L. The flaw tolerance evaluation, using the environmental

crack growth data in proposed ASME Code Case, "Fatigue Crack Growth Rate Curves for Ferritic Steels in PWR Water Environments," (Rev 1, 12/10/96), indicates that a postulated fatigue flaw in three of the steam generator feedwater nozzles could grow through wall in less than one operating cycle. The BGE license renewal application indicates that corrective actions will be initiated well in advance of reaching a fatigue usage factor of 1.0. Describe the corrective actions that will be initiated when the fatigue usage factor approaches 1.0 at the steam generator feedwater nozzles.

6. Section 5.9 of the application references a site report (Reference 33 on page 5.9-26 of the application) dated July, 1996, for the BGE fatigue evaluation. Other sections of the application reference this report or other apparently similar reports. In December, 1997, EPRI issued Report TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant." By letter dated February 9, 1998, EPRI submitted this report for staff information. Describe the extent to which EPRI Report TR-107515 is a current fatigue evaluation and the results of all of the other plant-specific fatigue analyses.

7. BGE's program for managing the effects of erosion/corrosion is to monitor the local pipe wall thickness and take corrective action when the wall thickness is projected to fall below a certain minimum value. Your July 30, 1998, response to the staff request for additional information (RAI) on the feedwater system, question 13, indicates that this minimum wall thickness is determined based on internal pressure alone.

   (a) Please demonstrate that piping with a pipe wall thinned locally to this minimum wall thickness could withstand all licensing basis loads, including bending.

   (b) The minimum wall thickness equation cited in your July 30, 1998, RAI response applies only to straight pipes. Please provide the basis for applying this equation to fittings, such as elbows, tees, reducers, and fabricated branch connections.

8. 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.

9. In order to measure the maximum wall thinning of a given component caused by erosion/corrosion, several measurements at different locations are made and the maximum wall thinning is calculated. Describe what approach is used for measuring data along a pipe (that is, band, area, moving blanket, or point to point method).

10. Describe the erosion/corrosion degradation of the feedwater check valves which was discovered during their inspection at the Calvert Cliffs plants. How was the inspection performed? Was the wall thinning measured or was the inspection limited only to visual examination?

11. In addition to the predictions by the CHECWORKS computer code, what other selection methods (for example, industry experience and engineering judgment) are used in

selecting components for erosion/corrosion inspection (wall thickness measurement)? Describe them briefly.

12. To determine the life of the components exposed to erosion/corrosion, it is important to know the rate at which thinning of their walls is occurring. This information can be obtained by using appropriate methods for trending component degradation due to erosion/corrosion. Describe the trending methods used in predicting life of the components. In your trending methods, are you using measured or computer predicted data?

13. What is the frequency of valve inspection in the Preventive Maintenance Program relied on to manage erosion/corrosion?

14. Describe the materials for replacement components in the feedwater system due to erosion/corrosion degradation, such as chromium-molybdenum and carbon steel.

15. 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.

16. Describe incidents of damage or failure of components caused by erosion/corrosion at Calvert Cliffs and associated corrective actions.

17. Does the BGE erosion/corrosion program permit weld overlay as a corrective action when degraded components are found?

18. Describe the extent of inspection of 2-inch and less piping as part of the BGE erosion/corrosion program.

September 1, 1998

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

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR THE SPENT FUEL POOL COOLING SYSTEM (TAC NOS. M99595, M99596, AND M99209)

Dear Mr. Cruse:

By letter dated August 21, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Spent Fuel Pool Cooling System (5.18) 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 (NRC) staff review report 5.18 to determine if the report meet 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 Spent Fuel Pool Cooling System (5.18) against the requirements of 10 CFR 54.21(a)(1), 10 CFR 54.21(a)(3). By letter dated April 4, 1996, the NRC 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 NRC 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 NRC 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
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Exhibit 4

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LDoerflein (LTD)
BBores (RJB)
SDroggitis (SCD)
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REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
MAIN STEAM, STEAM GENERATOR BLOWDOWN, EXTRACTION STEAM, AND NITROGEN AND HYDROGEN SYSTEMS
INTEGRATED PLANT ASSESSMENT, SECTION 5.12
DOCKET NOS. 50-317 AND 50-318

Section 5.12.1 Scoping

1. In Section 5.12.1 (bottom of Page 5.12-1) you identify that the Erosion Corrosion Program is credited for the mitigation of several components within the scope of license renewal. It is not clear what the term "mitigation of several components" is intended to imply. Please clarify this statement and explain its relationship to license renewal.

2. In Section 5.12.1 you also state that there have been problems with system drains associated with portions of the system not within the scope of license renewal (WSLR). It is not clear whether the system referred to is the extraction steam system, main steam system, steam generator blowdown system, or all three of these steam systems. Please clarify this statement.

3. The portions of the steam generator blowdown system (SGBS) that are inside containment are included (see Section 5.12.1) in the scope of Section 5.12. However, according to Section 5.12.1.1 the SGBS is apparently not included in the Section 5.12, nor does it appear to be included in any of the other referenced reports listed in Section 5.12.4. Please identify which section of the license renewal application (LRA) discusses scoping and aging management for the SGBS or provide the basis for its exclusion.

4. Section 5.12.1.1 describes the functional requirements of the main steam, extraction steam, and nitrogen and hydrogen systems, but does not provide similar information for the SGBS. Please describe the functional requirements of the SGBS. Also identify if there is a containment isolation function associated with the SGBS and indicate whether it is or should be included in this section of the License Renewal Application (LRA).

5. You have identified that the non-safety related portions of the main steam system that are WSLR for fire protection considerations are addressed in Section 5.10 of your application. The NRC staff would expect that the aging mechanisms and management programs would be the same for the non-safety and safety-related portions of the main steam system. In light of this assumption, provide a summary discussion on why portions of the main steam system are addressed in multiple sections of the LRA. Also, identify the functional requirements and intended functions of the main steam system that are within the scope of license renewal for fire protection considerations.

6. On Page 5.12-4, you identify that the main steam lines from the steam generators to the main steam isolation valves (MSIVs) are WSLR. Please clarify this statement to indicate if the scope includes the MSIVs and if the scope extends to the first restraint downstream of each MSIV. If it does not extend to the first restraint downstream of the

Enclosure

MSIV provide appropriate justification for the exclusion of this portion of the system (and its restraint).

7. You have identified that the extraction steam system inside containment has been abandoned in place and that only the associated containment penetration is WSLR. Assuming that the piping inside containment needs to be seismically supported for Seismic II over I considerations, describe whether the supports for this piping are WSLR. Additionally, please provide similar discussions for any other abandoned equipment that may potentially affect the performance of a safety-related function during a design basis event, and the extent to which that equipment was determined to be WSLR.

8. Section 5.12.1.2 identifies the air supply piping to the auxiliary feedwater (AFW) stop control valves as being WSLR. Describe why other air-operated components in the systems included in Section 5.12 are not WSLR. Also, explain why potential and plausible aging mechanisms (Table 5.12-4) for portions of the compressed air system that are included in this report are not identified, or provide an appropriate reference to where these components are subjected to an aging management review.

Section 5.12.2 Aging Management

9. Are there any parts of the systems structures or components described in Section 5.12 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. (1) Preventive actions that will mitigate or prevent aging degradation; (2) Parameters monitored or inspected relative to degradation of specific structure and component intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

10. Table 5.12-4 shows that fatigue is not plausible for the main steam system. However, Table 2.1-1 in Section 2.1 shows that the main steam piping fatigue is one of the time-limited aging analyses (TLAAs) that were determined to be subject to license renewal review. Additionally, in Section 2.1.3.4, a discussion is provided to demonstrate that the main steam piping fatigue analyses meet the criteria of 10 CFR 54.21 (c)(1)(I), such that the 7000 assumed thermal cycles will not be exceeded during the period of extended operation. Provide the basis for concluding fatigue is not a plausible age related

degradation mechanism in light of the above information. Please also discuss if there is an inconsistency between Tables 5.12-4 and 2.1-1 as a result of above information.

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 5.18.1 - Scoping

1. The simplified diagram of the spent fuel pool cooling system (SFPCS) on page 5.18-3 in subsection 5.18.1.1 shows system interfaces with the Solid Waste Disposal System and the Demineralized Water and Condensate Storage System. These interfaces show boundary valves (diaphragm valves) with small pipe segments extending a short distance beyond the boundary valve and then ending (no pipe support, isolation valve or apparatus is apparent). Are these small pipe segments within the scope of license renewal? If so, specify the interfaces at the end of these pipe segments that separate the portions of the system within the scope of license renewal from those portions of the system outside the scope of license renewal.

2. Calvert Cliffs Nuclear Power Plant (CCNPP) Updated Final Safety Analysis Report (UFSAR) Figure 9-7 (Baltimore Gas and Electric Company Drawing 64-314, Rev. 2) shows an additional interface (five piping connections to the demineralizer compared with the four shown on Figure 5.18-1) with the Spent Fuel Pool Demineralizer. The line appears to be from the Instrument Air System and is not addressed in Section 5.18. Please provide an evaluation of this line including its scoping boundaries for license renewal.

3. CCNPP UFSAR Figure 9-7 shows a portable resin addition tank connected to the spent fuel pool (SFP) Demineralizer by a spool piece. Figure 5.18-1 does not clearly indicate where the scoping boundary for the sections of piping up to and including the resin addition tank. Please provide the basis for why this portion of the spent fuel pool cooling system line including its boundaries were excluded from the scope of license renewal or a cross reference to where it is addressed in the license renewal application (LRA).

4. CCNPP UFSAR Figure 9-7 includes the following "device" that is not included in Table 5.18-1, "SFPCS Device Type Disposition" FG. Three of these devices are located in the piping connected to the demineralizer. Please explain what device type these symbols represent and how they are dispositioned for license renewal.

Section 5.18.2 - Aging Management

5. Section 5.18.1 indicates that there were several instances of cracking of SFPCS piping and a detailed study was performed in early 1990 to determine the root cause and appropriate remedy. The study determined that the cracking was due to high-cycle fatigue caused by cavitation-induced vibration. Subsequently, certain orifices and valves were modified to eliminate system cavitation. This section of the LRA also indicated that implementation of these improvements has prevented recurrence of cracking in SFPCS

piping. Please address whether the piping susceptible to cracking is subject to an aging management review (AMR). If so, please provide a summary discussion of the AMR performed for this piping that demonstrates there is reasonable assurance that the intended functions for these components will be maintained during the period of extended operation by managing high-cycle fatigue and other aging effects of the SFPCS piping. If not, provide the basis from excluding these components from an AMR.

Please address whether these modified orifices and valves are subject to an AMR. If so, please provide a summary discussion of the AMR performed for these orifices and valves that demonstrates there is reasonable assurance that the intended functions for these components will be maintained during the period of extended operation by managing high-cycle fatigue and other aging effects (e.g., erosion) of these orifices and valves. If not, provide the basis from excluding these components from an AMR.

Subsection 5.18.1.1 states that since normal service loads do not result in significant vibration or other dynamic loading conditions, fatigue is not plausible for SFPCS. Please provide the values and the basis for the determination of "significant vibration" and provide a description of the monitoring activities used to determine any post-modification vibration significance. Please indicate if monitoring is ongoing, performed periodically, or planned for some time in the future to indicate and track any future vibration.

The Calvert Cliffs UFSAR Section 9.4.3.2 states that the SFPCS piping was designed to ANSI B31.7 Code requirement. While the Code does not require an explicit fatigue analysis for Class II and III piping system, it does specify allowable stress levels based on the number of anticipated thermal cycles. Please provide a discussion on the fatigue evaluation for the SFPCS piping with respect to the requirements of 10 CFR 54.21(c)(1) focusing why the number of thermal cycles expected to occur during the period of extended operation will preclude reaching allowable stresses for the SFPCS components.

6. Table 5.18-3 indicates that wear is a plausible age-related degradation mechanism for hand valves in the SFPCS. Section 5.18.2 indicates that the local leak rate testing (LLRT) of the containment isolation requires corrective actions as part of the program which will ensure that the intended function of the containment isolation hand valves will be maintained under the current licensing basis during the period of extended operation.

   However, the staff noted in Figure 5.18-1 that the SFPCS includes certain hand valves that are not containment isolation valves and, therefore, are not subject to the LLRT. Please specify any aging management program for these valves to manage the effects of wear in order to maintain their intended function during the period of extended operation.

7. Table 5.18-4 in Section 5.18.2 shows the list of subcomponents and materials subject to aging. Provide the basis for excluding the SFP heat exchanger tubing, which has the intended function of removing heat from the SFP, the refueling pool water, and maintaining the pressure boundary of the SFPCS from this table.

intended function of removing heat from the SFP, the refueling pool water, and maintaining the pressure boundary of the SFPCS from this table.

8. Discuss plans for detection of inadvertent ingress of service water from the shell side of the heat exchanger which is at a higher pressure than that of the tube side, through degraded SFP heat exchanger tubes into spent fuel pool water and into other interface systems which could lead to a chloride excursion.

9. Are there any parts of the systems, structures and components within the SFPCS 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.

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.

Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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September 2, 1998

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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, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT ON METAL FATIGUE (TAC NOS. MA0601, MA0602, M99227, MA1016, MA1017, M99223, MA1108, MA1109, AND M99222)

Dear Mr. Cruse:

By letter dated April 8, 1998, Baltimore Gas and Electric Company (BGE) submitted its license renewal application. The NRC staff is reviewing the integrated plant assessment reports contained in the application against the requirements of 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(3). Based on a review of the information submitted, the staff has identified in the enclosure, areas regarding metal fatigue 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

Exhibit 5 DF01 1/1

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Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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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"

1. Section 5.2, page 5.2-14, of the application contains a list of Chemical and Volume Control System (CVCS) subcomponent parts for which fatigue is considered plausible. The application further indicates that the CVCS Charging Inlet Nozzle was identified as the most bounding location. Identify which subcomponents have fatigue analyses. Describe the review process used to evaluate the subcomponent parts for fatigue, including the selection of the bounding location.

2. Section 5.2 of the application indicates that the Fatigue Monitoring Program (FMP) tracks the fatigue usage at the Charging Inlet Nozzle. Describe the parameters that are monitored by the FMP that are applicable to the Charging Inlet Nozzle. Also describe how the monitored parameters are compared to the fatigue analysis of record.

3. Electric Power Research Institute (EPRI) Report TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant," dated December 1997, provides the results of the fatigue analyses of the CVCS piping. Section 3.2.1.1 of the EPRI report indicates that the existing fatigue analysis of the piping did not account for the auxiliary spray transients. The EPRI report further indicates that revised analyses are under development. Describe the manner by which the time-limited aging analyses (TLAA) for the revised CVCS fatigue analyses will satisfy 10 CFR 54.21(c) considering the existing analysis did not account for the auxiliary spray transients. Also provide the schedule for completion of the revised CVCS fatigue analyses. Also, describe the expected impact of these revised analyses on the evaluation contained in EPRI Report TR-107515.

4. Section 3.2.2.1 of EPRI Report TR-107515 indicates that the charging and auxiliary spray piping were reanalyzed to account for the installation of an orifice for the stop check valve in the bypass line around isolation valve CV-519. Section 3.2.2.3 of the EPRI report describes the back-projection of fatigue usage from FMP data, which was only available for the May through December 1995 time frame. Provide the date of the installation of the orifice in the bypass line. Describe the impact of the modification to the bypass line, if any, on the parameters monitored by the FMP. Also describe the impact of the modification, if any, on the computation of previous fatigue usage and the projection of fatigue usage to 40 and 60 years.

5. Section 3.2.2.5 of EPRI Report TR-107515 summarizes the fatigue cumulative usage factor (CUF) projections for the Charging System piping and Auxiliary Spray piping locations. The projected CUFs in these lines are higher than the projected CUF at the

CVCS Charging Inlet Nozzle. However, as discussed in item 1 above, the CVCS Charging Inlet Nozzle was identified as the most bounding fatigue location. Explain why the projected CUFs are higher in the Charging System and Auxiliary Spray piping locations than at the bounding location.

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.

   (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.

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

7. Section 5.2 of the application indicates that Calvert Cliffs Units 1 and 2 have experienced cases of fatigue failures in CVCS piping that were attributed to vibration loads imposed by operation of the Charging Pumps. The application indicates that BGE performed piping design modifications to reduce vibration and improve the CVCS reliability. Describe the modifications that were performed to reduce the vibration. Indicate whether vibration monitoring of the piping was performed subsequent to the modifications. Identify the Codes and Standards used, and summarize the significance of the results for the period of extended operation, if any, of the vibration monitoring.

8. To verify that no significant vibrational fatigue is occurring for the components, Section 5.2 of the application indicates that a new program will be developed to provide requirements for inspections of representative components. The application further indicates that the program details are discussed in the Aging Management Program section for CVCS Group 2. However, the Group 2 program is for managing the effects of corrosion. Discuss the specific elements of the Group 2 corrosion program that are relevant in monitoring vibration fatigue.

Section 4.1, "Reactor Coolant System"

9. Section 4.1 of the application indicates that Calvert Cliffs has shut down on several occasions due to Reactor Coolant System (RCS) leakage associated with the Reactor Coolant Pumps (RCPs). The application also indicates that a vibration monitoring and

reduction program has been implemented for the piping associated with the RCP seal leakoff lines. Describe the parameters that are currently monitored by this program. Also, provide the acceptance criteria for the monitored parameters including the technical basis for the acceptance criteria.

10. Section 4.1 of the application indicates that the FMP monitors and tracks low-cycle fatigue usage for the limiting components of the Nuclear Steam Supply System (NSSS) and Steam Generator (SG) safe-ends-to-reducer welds. Describe the parameters that are monitored by the FMP that are applicable to the NSSS and SG safe-end-to-reducer welds. Also describe how the monitored parameters are compared to the fatigue analysis of record.

11. Section 4.1 of the application indicates that a one-time fatigue analysis will be performed for the RCPs, Motor-Operated Valves (MOVs), and pressurizer relief valves to determine if these components are bounded by components and transients currently included in the FMP. Describe the fatigue criteria that were used in the original design of these components. Describe the purpose and criteria for the one-time fatigue analysis described in Section 4.1. Describe the manner by which the time-limited aging analyses (TLAA) for these fatigue analyses will satisfy 10 CFR 54.21(c). Also provide the schedule for completion of these fatigue analyses.

12. Section 4.1 of the application provides the CUFs through 1996 for the critical RCS components. Provide the projected CUFs for the critical RCS components at the end of the period of extended operation.

13. Section 4.1 of the application indicates that in order to remain within the design basis, corrective action is initiated well in advance of the CUF approaching one or the number of cycles approaching design allowable. Describe the specific criteria used to determine when corrective actions will be initiated.

14. EPRI Report TR-107515 provides the results of a fatigue assessment of the Pressurizer Surge Line. Section 3.3.1.1 of the EPRI report provides the results of an ASME Code Section III evaluation of the line that had been performed to address the issue of fatigue due to thermal stratification. The EPRI report lists a Class 1, Equation 12 stress that exceeds the ASME Code allowable limit. No further explanation is provided. Indicate whether the ASME Code stress limits were met for this analysis.

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).

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).

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.

Section 5.15, "Safety Injection System"

18. Section 5.15 of the application contains a list of Safety Injection (SI) System components for which fatigue is considered plausible. The application indicates that the SI System vent/drain/test hand valves, instrument isolation hand valves, and relief valves connected to the piping are generally "thin-walled" components and, therefore, do not experience the large temperature gradients that would be necessary to cause significant degradation. Provide the technical basis for this conclusion.

19. Section 5.15 of the application indicates that the FMP tracks the fatigue usage at the SI Nozzle. Describe the parameters that are monitored by the FMP that are applicable to the SI Nozzle. Also describe how the monitored parameters are compared to the fatigue analysis of record.

20. Section 5.15 of the application indicates that in order to stay within the design basis, corrective action is initiated well in advance of the CUF approaching one or the number of cycles approaching the design allowable. Describe the specific criteria used to determine when corrective actions will be initiated.

21. Section 5.15 of the application indicates that BGE identified the potential for thermal stratification in the piping between the SI Tank check valves and the loop inlet check valves. The application also indicates that BGE will complete an engineering review of the industry task group reports regarding thermal stratification to determine whether SI piping changes are necessary, and to determine the impact of such changes on fatigue usage parameters used by the FMP. Indicate whether the plans for the engineering review includes reanalysis for thermal stratification. Describe the manner by which the time-limited aging analyses (TLAA) for these fatigue analyses will satisfy 10 CFR 54.21(c). Also provide the schedule for completion of these fatigue analyses.

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).

COMMENTS ON THE APPLICATION OF THE EPRI ENVIRONMENTAL FATIGUE FACTOR TO THE CALVERT CLIFFS PLANTS

The environmental factor approach described in the report is a convenient and acceptable method to incorporate the effects of LWR coolant environments on fatigue life of pressure vessel and piping steels. However, the correlations for calculating the fatigue life correction factors F_{en} should be updated. For carbon and low–alloy steels, the dependence of F_{en} on dissolved oxygen (DO) is not consistent with experimental data. For austenitic stainless steels, the correlations do not include the effects of DO content and temperature; particularly the effects of DO content are important because environmental effects are more pronounced in low–DO PWR environments than in high–DO water.

Another minor point, the report makes several references to the fact that environmental factor approach gives a lower usage factor than the interim fatigue design curves of NUREG/CR–5999, implying that this difference is due to the methodology, i.e., graphical versus mathematical representation of the best–fit curve of the experimental data. The methodology will introduce a difference if the best–fit curves used in developing the current Code design fatigue curves are different than the best–fit curves of the present fatigue S–N data, because the design curves not only account for the effects of environment but also small differences that might exist between the ASME mean curve and the best–fit curve of existing fatigue data.

For carbon and low–alloy steels, because the ASME mean curves are either comparable or somewhat conservative, the two methods should yield similar results as long as the same correlations are used in developing the design curve and the correction factors. Minor differences between the two mentioned in this report are due to the correlations used for the interim curves. For austenitic stainless steels, it is well known (Jaske & O'Donnell, 1978) that the ASME mean curve is inconsistent with the existing fatigue data. Experimental fatigue lives are a factor of up to 3 lower than those predicted by the ASME mean curve. Consequently, usage factors based on interim design curves may be significantly higher because they account for this difference. However, for austenitic stainless steels, the margin factors on life are lower than 20 and closer to 10 or 8, i.e., there is little or no safety margin to account for environmental effects. Some specific comments on the report are as follows.

SECTIONS 2.2.3 & 3.1.3: ENVIRONMENTAL EFFECTS

The report follows the methodology of EPRI TR–105759, "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations," to account for the effects of reactor coolant environment on the fatigue life of components. This approach was initially proposed by Higuchi and Iida (1991). The effects of coolant environment on fatigue life are expressed in terms of a fatigue life correction factor F_{en}, which is the ratio of the life in air at room temperature to that in water at the service temperature. This method is also being proposed as a non–mandatory Appendix.

To incorporate environmental effects into the ASME Code fatigue evaluation, a fatigue usage for a specific load pair based on the current Code design curve is multiplied by the correction factor. The correlations for F_{en} are based on the statistical models developed by ANL (NUREG/CR–6335, 1995). The statistical models have since been updated. The models for carbon and low–alloy steels were first modified (Gavenda et. al. PVP-Vol. 350, 1997) because it was determined that in the range of 0.05 to 0.5 ppm, the effect of DO was more logarithmic than linear. Recently, these models have been further optimized with a larger data base (Chopra & Shack PVP 98; also NUREG/CR–6583, 1998). The models in NUREG/CR–6335 for austenitic stainless steels were based on very limited data, and have also been updated to incorporate the effects of DO, temperature, and strain rate on fatigue life (Chopra & Smith, PVP 98). These updated models should be used to estimate F_{en} in LWR environments.

In addition, a set of threshold values of strain amplitude, strain rate, temperature, dissolved oxygen (DO), and sulfur content are defined for environmental effects to occur. In NUREG/CR–6335, these threshold values were defined on the basis of experimental observations and trends in the existing fatigue S–N data. With the exception of strain amplitude, the same threshold values have been included in the non–mandatory Appendix. A threshold strain amplitude of 0.1% is proposed for both carbon and low–alloy steels as well as austenitic stainless steels in the Appendix; the basis for this value is not provided. The threshold strain should be related to the rupture strain of the surface oxide film; there is little data to establish this value. Limited data suggest that for carbon and low–alloy steels, the threshold strain is ≈20% higher than the fatigue limit of the steel (i.e., ≈0.11 and 0.15%, respectively, for carbon steels and low–alloy steels). A threshold strain amplitude of 0.16% has been observed for austenitic stainless steels. Unless it can be demonstrated otherwise, these values must be adjusted for the effects of mean stress and uncertainties due to material and loading variability, which yields threshold strain amplitude of 0.07% (21 ksi or 145 MPa) for carbon and low–alloy steels and 0.097% (27.5 ksi or 189 MPa) for stainless steels.

The EPRI report TR–105759 gives a different set of threshold values that represent the strain rate, temperature, and DO level which results in "moderate" or "acceptable" effects of environment, i.e., a factor of up to 4 decrease in fatigue life. For example, environmental effects on life for 0.1 ppm DO level are considered acceptable, and F_{en} is considered to be 1. Although a factor of 3 or even 4 on life appears reasonable for carbon and low–alloy steels (Chopra & Shack, PVP 98), the EPRI report does not provide a technical basis for selecting a factor of 4 as a working definition of acceptable effects. However, this approach results in a discontinuity at the threshold value, e.g., F_{en} is 1 at 0.1 ppm DO and may jump to 10 or higher at 0.105 ppm. To avoid such discontinuities, experimental threshold values (e.g., NUREG/CR–6583) should be used to determine F_{en}; then to take advantage of the conservatism in design fatigue curves, the calculated values may be divided by 3. In other words, up to a factor of 3 decrease in life due to environment is ignored in the evaluations. This approach is being considered by EPRI.

Please note that the above approach (factor of 3 decrease in life being acceptable) is applicable only for carbon and low–alloy steels and not for austenitic stainless steels. The reason being that the current ASME Code mean curve for low–alloy steels is consistent with the existing fatigue S–N data and that for carbon steels is somewhat conservative. Thus, a factor 3 margin on life may be used to account for acceptable effects of environment. However, the current ASME Code mean curve for austenitic stainless steels are not consistent with the existing fatigue S–N data; a margin of only 10 on life and 1.5 on stress exists

between the Code design curve and the mean curve (Chopra & Smith, PVP 98). Consequently, a factor of less than 1.5 margin on life may be used to account for acceptable effects of environment.

EXECUTIVE SUMMARY (PAGE 2, "RESULTS")

".... application of the effects of reactor water environments, ... produces worst–case environmental multipliers that are already compensated for by two existing conservatisms in Class 1 ASME Code fatigue analysis procedures – (1) the low–cycle portion of the design fatigue curve margin factor of 20 that is appropriately ascribed to moderate environmental effects, and .... "

Please note that the factors of 20 on life and 2 on stress should not be considered as safety margins but rather conversion factors that must be applied to the experimental data to obtain reasonable estimates of the lives of actual reactor components. Although in a benign environment some fraction of the factors, e.g., a factor of 3 on life, may be available as a safety margin.

Also, fatigue tests conducted on 0.914 m (36 in.) diameter vessels with 19 mm (0.75 in.) wall in room–temperature water at Southwest Research Institute for the Pressure Vessel Research Council (Kooistra, et al., 1964) show that ≈5 mm deep cracks can form in carbon and low–alloy steels very close to the values predicted by the ASME Code design curve. These results demonstrate clearly that the Code design fatigue curves do not necessarily guarantee any margin of safety.

MEETING WITH ELECTRIC POWER RESEARCH INSTITUTE ON METAL FATIGUE, MARCH 19, 1998

The methodology and results from four studies on Environmental Fatigue Evaluations, e.g., Calvert Cliffs, Older Westinghouse Plants, Representative BWR Components, and Newer Vintage BWR Plants, were presented at the meeting. All studies essentially follow the environmental factor approach described in the EPRI report TR–105759, and used in the EPRI report TR–107515 on evaluation of thermal fatigue effects for Calvert Cliffs Nuclear Power Plant.

The effects of coolant environment on fatigue life are expressed in terms of a fatigue life correction factor or environmental factor F_{en}, which is the ratio of the life in air to that in water. A fatigue usage for a specific load pair based on the current Code design curve is multiplied by the correction factor. The correlations for F_{en} are based on the statistical models developed by ANL (NUREG/CR–6335, 1995), which also include a set of threshold values of strain amplitude, strain rate, temperature, and dissolved oxygen beyond which environmental effects on fatigue life are significant. A detailed description of the EPRI methodology is given below.

Correlations Based on NUREG/CR–6335

F_{en} for carbon steels (CSs) and low–alloy steels (LASs) is expressed as

CSs $$F_{en} = \exp(0.384 - 0.00133\,T - 0.554\,S^*\,T^*\,\dot{\varepsilon}^*\,O^*) \quad (1)$$

LASs $$F_{en} = \exp(0.766 - 0.00133\,T - 0.554\,S^*\,T^*\,\dot{\varepsilon}^*\,O^*), \quad (2)$$

where the threshold and saturation values (the value beyond which the effect of environment saturates) of sulfur content S, temperature T, strain rate $\dot{\varepsilon}$, and DO content in water are defined as

$$S^* = S \quad (0 < S \le 0.015 \text{ wt.\%})$$
$$S^* = 0.015 \quad (S > 0.015 \text{ wt.\%}) \quad (3a)$$

$$T^* = 0 \quad (T < 150°C)$$
$$T^* = T - 150 \quad (T \ge 150°C) \quad (3b)$$

$$\dot{\varepsilon}^* = 0 \quad (\dot{\varepsilon} > 1\%/s)$$
$$\dot{\varepsilon}^* = \ln(\dot{\varepsilon}) \quad (0.001 \le \dot{\varepsilon} \le 1\%/s)$$
$$\dot{\varepsilon}^* = \ln(0.001) \quad (\dot{\varepsilon} < 0.001\%/s) \quad (3c)$$

$$O^* = 0 \quad (DO < 0.05 \text{ ppm})$$
$$O^* = DO \quad (0.05 < DO \le 0.5 \text{ ppm})$$
$$O^* = 0.5 \quad (DO > 0.5 \text{ ppm}) \quad (3d)$$

F_{en} for Types 304 and 316 stainless steels (SSs) is expressed as

$$F_{en} = \exp(0.359 - 0.134\ \dot{\varepsilon}^*) \tag{4}$$

where the threshold and saturation values of strain rate $\dot{\varepsilon}$ are defined as

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 1\%/s) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}) && (0.001 \le \dot{\varepsilon} \le 1\%/s) \\ \dot{\varepsilon}^* &= \ln(0.001) && (\dot{\varepsilon} < 0.001\%/s) \end{aligned} \tag{5}$$

Updated Correlations for Fatigue Life in LWR Environments

The models for CSs and LASs were later updated (PVRC Meeting, Orlando, April 1996) because the existing fatigue S–N data indicate that in the range of 0.05–0.5 ppm, the effect of DO on life (Eq. 3d) was more logarithmic than linear. Thus, updated correlations of F_{en} for CSs and LASs are expressed as

$$\text{CSs} \qquad F_{en} = \exp(0.384 - 0.00133\ T - 0.1097\ S^*\ T^*\ \dot{\varepsilon}^*\ O^*) \tag{6}$$

$$\text{LASs} \qquad F_{en} = \exp(0.766 - 0.00133\ T - 0.1097\ S^*\ T^*\ \dot{\varepsilon}^*\ O^*), \tag{7}$$

where the threshold and saturation values of sulfur content S, temperature T, and strain rate are the same as those defined in Eqs. 3a–3c, and those of DO content are defined as

$$\begin{aligned} O^* &= 0 && (DO < 0.05\ ppm) \\ O^* &= \ln(DO/0.04) && (0.05 < DO \le 0.5\ ppm) \\ O^* &= \ln(12.5) && (DO > 0.5\ ppm) \end{aligned} \tag{8d}$$

These correlations (Eqs. 6 and 7) have been further optimized with a larger data base (Chopra & Shack, PVP 1998). The differences between the optimized correlations and Eqs. 6 and 7 are minimal; the differences are essentially in estimates of life in low–DO environments.

The NUREG/CR–6335 models for austenitic SSs (Eqs. 4 and 5) were based on very limited data. For example, nearly all of the data in water were obtained at high temperatures (280–320°C) and high levels of DO (0.2–8 ppm). The data were inadequate to establish the dependence of life on strain rate, temperature, or DO content, or to define the threshold and saturation values of these parameters. These models have now been updated with a larger data base (Chopra & Smith, PVP 1998). The updated correlation of F_{en} for Types 304 and 316 SS is expressed as

$$F_{en} = \exp(0.935 - T^*\ \dot{\varepsilon}^*\ O^*) \tag{9}$$

where the threshold and saturation values of temperature T, strain rate $\dot{\varepsilon}$, and DO content in water are defined as

$$\begin{aligned} T^* &= 0 && (T < 200°C) \\ T^* &= 1 && (T \ge 200°C) \end{aligned} \tag{10a}$$

$$\dot{\varepsilon}^* = 0 \qquad (\dot{\varepsilon} > 0.4\%/s)$$
$$\dot{\varepsilon}^* = \ln(\dot{\varepsilon}/0.4) \qquad (0.0004 \le \dot{\varepsilon} \le 0.4\%/s)$$
$$\dot{\varepsilon}^* = \ln(0.0004/0.4) \qquad (\dot{\varepsilon} < 0.0004\%/s) \qquad (10b)$$

$$O^* = 0.260 \qquad (DO < 0.05 \text{ ppm})$$
$$O^* = 0.172 \qquad (DO \ge 0.05 \text{ ppm}) \qquad (10c)$$

Please note that F_{en} is greater in low–DO PWR than in high–DO environments.

The EPRI Environmental Factor Approach

1) Because the current fatigue design curves are based on data obtained in room–temperature air, an environmental correction factor should be determined with respect to room–temperature air, i.e., F_{en} should be defined as ratio of the life in air at room temperature to that in water at the service temperature. It will retain the margins of 20 on life and 2 on stress that are used to develop design fatigue curves from the best–fit experimental curves. In the EPRI approach, F_{en} is defined as ratio of the life in air to that in water both at the service temperature. The premise being that the effect of environment alone needs to be incorporated in F_{en}; margins of 20 and 2 in the current design curves are adequate to account for the uncertainties that arise due to other factors.

2) The correlations for F_{en} are based on the statistical models of NUREG/CR–6335 (Eqs. 1, 2, and 4). As discussed above, F_{en} should be determined from the updated correlations (Eqs. 6, 7, and 9).

3) In EPRI report TR–105759, a different set of threshold values (other than Eqs. 3, 8, and 10) are defined such that they result in "moderate" or "acceptable" effect of environment (i.e., they result in up to a factor of 3 decrease in fatigue life). For example, when all other threshold conditions are satisfied, a DO level of 0.1 ppm may result in a factor of 3 decrease in life. Therefore, a threshold value of 0.1 ppm DO is used in the evaluations, i.e., F_{en} is 1 for all load pairs with ≤0.1 ppm DO. Although a factor of 3 on life appears reasonable for defining moderate or acceptable effects of environment on life of CSs and LASs, it can not be used for austenitic SSs. The existing fatigue S–N data for austenitic SSs indicate that the difference between the ASME Code design curve and best–fit experimental curve is closer to margins of 10 on life and 1.5 on stress than the 20 and 2 originally intended. Also, care should be taken to avoid taking credit for this factor twice, e.g., after eliminating all load pairs that do not satisfy the modified thresholds, a factor of up to 3 increase in CUF may be considered as "acceptable" effect of environment.

4) The existing fatigue S–N data can not justify a threshold value of 0.1% for strain amplitude, particularly for CSs and LASs.

September 2, 1998

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR THE REACTOR COOLANT SYSTEM (TAC NOS. MA1016, MA1017, AND M99223)

Dear Mr. Cruse:

By letter dated December 17, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Reactor Coolant System (4.1) 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 (NRC) staff review the Reactor Coolant System (4.1) report to determine if this 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 Coolant System (4.1) 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 related to scoping is needed to complete its review. Should the staff have additional information needs related to aging management they will be forwarded under a future correspondence.

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
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Exhibit 6

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SDroggitis (SCD)
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RLatta (RML1)
EHackett (EMH1)
AMurphy (AJM1)
TMartin (TOM2)
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GMeyer (GWM)
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SStewart (JSS1)
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SDroggitis (SCD)
DSolorio (DLS2)
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TCollins (TEC)
RCaruso (RXC)
MRazzaque (MMR1)
SLittle (SLL)

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

Section 4.1.1 - Scoping

Regarding the structures and components identified as being within the scope of license renewal for the Reactor Coolant System (RCS), the NRC staff has the following questions:

1. Please explain why the pressurizer spray and auxiliary spray nozzles were not included within the scope of license renewal?

2. In Table 4.1-2, "Tank (TK)" was listed as a device type requiring aging management review (AMR). But, Figure 4.1-1 shows that the Quench Tank No. 11 is not within the scope of license renewal. Please clarify this apparent discrepancy.

3. The device type, "Miscellaneous (XL)," listed in Table 4.1-1 has been classified as only associated with active functions, and therefore, was excluded from the AMR. Please describe the types of components that make up this device type.

4. In Table 4.2-2 in Section 4.2, footnotes were used to indicate that "not all components of a device type were affected by the ARDM." This has been interpreted to mean that some components within the device type category are not subject to the effects of the listed plausible aging related degradation mechanism (ARDM). Referring to Table 4.1-3 in subsection 4.1.1.2, please clarify whether any subcomponents of the components listed in the table are similarly not subject to the plausible ARDMs shown.

Enclosure

50-317
P

September 2, 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, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT REPORTS FOR THE CONTAINMENT ISOLATION GROUP, CONTAINMENT SPRAY SYSTEM, AND PRIMARY CONTAINMENT HEATING AND VENTILATION SYSTEM (TAC NOS. MA0603, MA0604, M99211, MA1038, MA1039, M99221, MA1106, MA1107, AND M99224)

Dear Mr. Cruse:

By letters dated November 14, 1997, January 21, 1998, and March 3, 1998, Baltimore Gas and Electric Company (BGE) submitted for review the Containment Isolation Group (5.5), Containment Spray System (5.6), and Primary Containment Heating and Ventilation System (5.11B) integrated plant assessment technical reports, respectively, as attached to the "Request for Review and Approval of System and Commodity Reports for License Renewal." BGE requested that the Nuclear Regulatory Commission (NRC) staff review reports 5.5, 5.6, and 5.11B to determine if these reports meet 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 reports 5.5, 5.6, and 5.11B 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 NRC staff has identified in the enclosures, 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 NRC 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

Exhibit 7

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: Request for Additional Information (3)
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BBores (RJB)
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REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
CONTAINMENT ISOLATION GROUP
INTEGRATED PLANT ASSESSMENT, SECTION 5.5
DOCKET NOS. 50-317 AND 50-318

Section 5.5.1 - Scoping

1. Clarify whether all the containment isolation valves listed in Table 5-3, "Containment Isolation Valves," of the Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report subject to an aging management review. For any valves that are not, provide the basis for their exclusion.

Section 5.5.2 - Aging Management

2. In Groups 1 and 2 under aging management programs and demonstration of aging management, the statement is made that the occurrence of crevice corrosion, general corrosion, microbiologically induced corrosion, and pitting is expected to be limited and not likely to affect the intended function of the Group 1 and 2 components. Provide the basis for this conclusion.

3. ASME Code Section III, ANSI B31.1 and ANSI B31.7 contain certain fatigue analysis requirements. For ASME Code Class 1 components and ANSI B31.7 piping, the Code requires the calculation of the cumulative usage factor. For ASME Code Class 2 and 3 components, and ANSI B31.1 piping, the Code specifies allowable stress levels based on the number of anticipated transients or thermal cycles. Explain why, in Table 5.5-2, fatigue is not considered as a plausible aging mechanism for the containment isolation (CI) group components, which are designed in accordance with ANSI B31.7 or similar requirements of ASME Code Section III.

4. ASME Code Section XI requires system leakage tests and system hydrostatic tests along with certain visual inspections for Class 2 and 3 components. Describe, in summary form, how these Section XI requirements are applied to CI group components.

5. Are there any parts of the systems, structures, or components described in Section 5.5 that are inaccessible for inspections? 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: (1) Preventive actions that will mitigate or prevent aging degradation; (2) Parameters monitored or inspected relative to degradation of specific structure and component

intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
CONTAINMENT SPRAY SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.6
DOCKET NOS. 50-317 AND 50-318

Section 5.6.1 - Scoping

1. Section 6.4.2 of the Calvert Cliffs Nuclear Power Plant (CCNPP) Updated Final Safety Analysis Report (UFSAR) states that "It is expected the containment spray will be effective in removing fission products from the containment atmosphere." Discuss why this intended function is not included as part of the system description or the system scoping results in Section 5.6 of the license renewal application (LRA). If this intended function is included, describe the components included within the scope of license renewal and subject to an aging management review. If not, justify why this function is excluded.

2. Discuss why the shutdown cooling intended function, as described in the CCNPP UFSAR is not included as one of the system scoping results in Section 5.6.1.1 of the LRA. If this intended function is included, describe the components included within the scope of license renewal and subject to an aging management review. If not, justify why this function is excluded.

3. Provide the basis for excluding spray nozzles shown in Figure 5.6-1 in Section 5.6.1.1 from the scope of license renewal.

4. Chapter 6 of the CCNPP UFSAR states that the containment spray system supplies the emergency dousing nozzles for the iodine removal units. The ability to put out charcoal fires due to decay heat from buildup of fission products, is normally relied upon at some nuclear power plants as an emergency dousing function. Provide the basis for not including the ability of the containment spray system to supply the emergency dousing nozzles for the iodine removal units as an intended function in Section 5.6.

Section 5.6.2 - Aging Management

5. Are there any parts of the systems, structures, or components described in Section 5.6 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: (1) Preventive actions that will mitigate or prevent aging degradation; (2) Parameters

monitored or inspected relative to degradation of specific structure and component intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
PRIMARY CONTAINMENT HEATING AND VENTILATION SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.11B
DOCKET NOS. 50-317 AND 50-318

Section 5.11B.1 - Scoping

1. Section 5.11B.1.2 of the LRA states that the portion of the Containment Air Recirculation and Cooling System within scope includes: cooling units, fans, and connecting ductwork up to and including the fusible dropout plates. Section 6.5.5, "Containment Air Recirculation and Cooling System," of the Calvert Cliffs Nuclear Power Plant (CCNPP) Updated Final Safety Analysis Report (UFSAR) states that each fan discharge duct is provided with a fusible link door that opens at an abnormally high containment temperature such as would occur under a loss-of-coolant accident. While Section 6.5.6 of the CCNPP UFSAR also states that the containment air cooler blowdown door fusible links are to be replaced every refueling outage to ensure that the links perform their design function and as a result would not be subject to an aging management review, clarify on what basis were the fusible links excluded from the scope of license renewal.

2. Section 6.5.6, "Containment Air Recirculation and Cooling System," of the UFSAR concludes that water-logging of the cooling units' coils is not a problem because the coil section drainage characteristics were validated by the manufacturer's sizing and test program. For this conclusion to remain valid, the staff believes that to drain condensate would have to be an intended function of the system. If it is an intended function of the system, clarify whether the piping described in Section 6.5.4 of the UFSAR which transfers the condensate leaving the coils to the containment sump and ultimately to the waste processing system is within the scope of license renewal and subject to an aging management review? If not, justify why this function is excluded.

3. Clarify whether the instrument lines are included in the scope of license renewal. 10 CFR 54.21(a)(1)(i) excludes instrumentation from the scope of renewal, in part because the instruments are routinely subjected to surveillance testing. The sample lines to such instruments as pressure transmitters, pressure indicators, water level indicator, and containment atmosphere draw samples (like those described in Section 6.8 of the UFSAR, "Hydrogen Control Systems," are not always tested to the same extent as the associated instruments. If the instrument lines have been excluded from the scope of license renewal, provide the justification for that exclusion with consideration of the foregoing concern.

4. Section 6.8.2, "Electric Hydrogen Recombiner," of the CCNPP UFSAR states that the service life of the recombiners is 40 years. Describe how this component was addressed for license renewal.

5. Section 5.11B.1.3 of the LRA states that the hydrogen recombiner only functions actively. This appears to be inconsistent with Section 6.8.2.3 of the CCNPP UFSAR which states that the recombiner is a completely passive device. Because the recombiner housing acts as a passive holdup volume to allow the containment atmosphere to be heated to a temperature above 1150°F, please provide the basis for considering the hydrogen recombiner to only have active functions and therefore not subject to an aging management review.

Section 5.11B.2 - Aging Management

6. Are there any parts of the systems, structures, or components described in Section 5.5 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: (1) Preventive actions that will mitigate or prevent aging degradation; (2) Parameters monitored or inspected relative to degradation of specific structure and component intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

50-317
P

September 3, 1998

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR THE REACTOR COOLANT SYSTEM (TAC NOS. MA1016, MA1017, AND M99223)

Dear Mr. Cruse:

By letter dated December 17, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Reactor Coolant System (4.1) 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 (NRC) staff review report 4.1 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 report 4.1 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
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Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
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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

1. Table 4.1-2 of the application indicates that Reactor Coolant System (RCS) piping with "device codes" of "-CC," "-GC," "-HB," and "-HC" are subject to aging management review (AMR). Please explain these "device codes" and describe components represented by them. Also, the description should identify whether these components include cold-leg, hot-leg, pressurizer surge line, spray line, connected American Society of Mechanical Engineers (ASME) Class 1 branch lines, and nozzles and safe ends at the reactor vessel, steam generators, pressurizer, pumps, and valves.

2. Provide a summary of the RCS piping sizes, piping material, and the corrosion allowances used in the design. Describe the basis upon which Baltimore Gas and Electric Company (BGE) concluded that the corrosion allowances are adequate for the period of extended operation.

3. The application does not apparently discuss several aging effects associated with certain RCS components. Summarize how the following aging effects have been addressed by BGE's aging management review.

   a. crack initiation and growth (stress corrosion cracking (SCC)) for the pressurizer shell/heads (including clad cracking), spray line nozzle, surge line nozzle, valve nozzle, manway, support skirt, integral attachments, and Unit 2 heater sleeve;

   b. corrosion and boric acid wastage for the pressurizer instrument nozzle and integral attachments;

   c. loss of preload for the pressurizer manway bolting.

   d. crack initiation and growth (SCC) for the RCS carbon steel (c/s) – hot and cold leg piping, nozzles, safe ends, and integral support;

   e. SCC for stainless steel (s/s) – reactor coolant pump (RCP) nozzles, safety and relief valve bodies and body flanges, bonnet and bonnet flanges, and nozzles; hot and cold leg, surge line, spray line, nozzles and safe ends; for s/s auxiliary piping of the decay heat removal system, core flood system and any other included Class 1 piping; fittings, nozzles, and safe ends of auxiliary piping; and component integral supports; cast austenitic stainless steel (CASS) – RCP casing, cover, casing flange, cover flange; safety and relief valve bodies, bonnets, body and bonnet flanges; cold and hot legs; surge line, nozzles; fittings, nozzles, and safe ends of auxiliary piping;

   f. SCC for nickel alloy – auxiliary piping safe ends;

Enclosure

g. SCC for High strength low alloy (HSLA) steel – RCP closure bolting and safety valves closure bolting;

h. general corrosion (boric acid corrosion from leakage of primary coolant) for integral supports (c/s), safety and relief valve bodies, bonnets, body flange, bonnet flange (s/s and CASS); RCP casing, cover, casing flange, cover flange (CASS); and safety valve closure bolting;

i. thermal embrittlement for CASS components – RCP casing and cover flanges; safety and relief valve body, bonnet, body and bonnet flange, hot and cold legs; surge lines; nozzles and safe ends; auxiliary piping fittings, nozzles, and safe ends;

j. loss of preload/stress relaxation for RCP closure bolting and safety and relief valve closure bolting.

4. The application does not apparently contain an AMR of the following pressurizer components: heater belt forgings; heater sheaths and end caps; heater bundles; and bundle cover plates. If these components are applicable to the Calvert Cliffs units, describe where these components are addressed in the LRA, or justify why these components have been excluded.

5. 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.

6. Describe the manner by which Procedure STP-M-574-1/2, "EC Examination of CCNPP ½ Steam Generators," manages aging effects.

7. How is erosion/corrosion managed for the secondary manway and cover plate, hand hole and cover plate?

8. 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.

9. Steam generator tubes have experienced intergranular attack (IGA). The application does not identify IGA as an aging issue. Provide basis for this determination.

10. Discuss how BGE will manage SCC of the CASS surge nozzle safe end.

11. What are the acceptance criteria in Procedure RV-78, "RV Flange Protection Ring Removal and Closure Head Installation?"

12. Describe how denting and pitting of the SG tubes will be managed.

13. 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 ½ 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"

14. Clarify whether crevice corrosion for the RCS is a plausible aging effect and, if so, provide a reference to where aging management is addressed in the LRA. If crevice corrosion is not a plausible aging effect for the RCS, describe the basis for that conclusion.

15. The application discusses prior degradation of the RCP suction deflector at Calvert Cliffs. What was the cause of the suction deflector bolting failures? What was the material of the bolts that failed, and how are the bolts being managed for aging?

16. Are there any parts of the systems, structures and components within the RCS 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.

50-317

September 3, 1998

Mr. Charles H. Cruse, Vice President
Nuclear Energy Division
Baltimore Gas & 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, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR REACTOR VESSEL INTERNALS SYSTEMS (TAC NOS. M98835, M98837, AND M99181)

Dear Mr. Cruse:

By letter dated May 23, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Reactor Vessel Internals System (4.3) 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 (NRC) staff review report 4.3 to determine if the report meet 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 report 4.3 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

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
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Mr. Charles H. Cruse
Baltimore Gas & Electric Company

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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 4.3.1 - Scoping

1. Figure 3.3-6 (Rev. 21) of the Calvert Cliffs Nuclear Power Plant (CCNPP) Updated Final Safety Analysis Report (UFSAR) shows the fuel assembly hold down (FAHD) structure. One of the intended functions of FAHD is to prevent fuel assemblies from being lifted out of position under accident loading conditions. Please clarify whether the FAHD was subjected to an aging management review (AMR), particularly the springs in it, which may loose their required force at extended age.

2. Figure 3.3-14 (Rev. 21) of the CCNPP UFSAR shows the upper guide structure (UGS) Assembly. Please describe the functions of the Expansion Compensating Ring, and indicate if its intended functions would meet the definition of intended function listed in 10 CFR 54.4(a).

3. Section 4.1.3.6 (Rev.18) of the CCNPP UFSAR indicates that vents were added to the reactor vessel and to the pressurizer head in response to the Three Mile Island Lessons Learned Report, NUREG-0737, Item II.B.1. One of the intended functions of the vents is to ensure core cooling during loss-of-coolant accident. Please indicate if this vent system was subjected to an AMR. If so, provide a cross reference to where the vents are addressed in the license renewal application (LRA). If not, provide the basis for their exclusion.

Section 4.3.2 - Aging Management

4. Clarify whether all the reactor vessel internal (RVI) components listed in Table 4.3-1 are within the scope of the ASME Code, Section XI, Subsection IWB inservice inspection program, as mentioned in Page 4.3-12. In addition, describe the applicable acceptance criteria and describe the methods used for trending for the visual inspection.

5. The aging management programs for Group 5 (Stress relaxation) described starting on page 4.3-24 indicate that plant-specific analysis will be performed to refine the calculated stress levels on control element assembly (CEA) shroud bolts and core shroud tie rods and bolts for verifying low tensile stress during normal operations, and for justifying no loss of preload due to stress relaxation. Provide the acceptance criteria that will be used for this analysis, and the schedule for completion of the analysis.

6. Page 4.3-24 indicates that an examination of the CEA shroud bolts and core shroud tie rods and bolts would be conducted as a part of an age related degradation inspection (ARDI) program if the refined stress level does not show the low stress expected. Assuming the results did warrant an ARDI for these components, provide a summary

Enclosure

discussion of the ARDIs consistent with the NRC staff's request for additional information on ARDIs in letter dated August 28, 1998, "Request for Additional Information For the Review of the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 & 2, Integrated Plant Assessment Report."

7. Section 4.3.1.1 indicates Section 3.3.3 of the UFSAR provides a description of the RVI structures. Section 3.3.3 does not provide sufficient details of the RVI components identified in Table 4.3-1 from Section 4.3.1. Please provide diagrams that show the location of the device types identified in Table 4.3-1.

8. Do the RVI intended functions include: (a) support for the irradiation surveillance capsules, and (b) shielding for the reactor pressure vessel (RPV)? If so, summarize what components perform these intended functions and explain whether these components are within the scope of license renewal.

9. CCNPP license renewal application addresses certain applicable aging effect for specific reactor vessel internals components. Describe, in summary form, the extent to which the following aging effects were determined to be either non-plausible or non-potential, for the specific components: stress corrosion cracking (SCC) and irradiation assisted stress corrosion cracking (IASCC), corrosion for the upper guide structure support plate; SCC, IASCC, corrosion, and wear for the control element assembly (CEA) shrouds; IASCC and corrosion for the CEA shroud bolts; SCC, IASCC, corrosion, and wear for the fuel alignment plate; SCC, IASCC, corrosion for the core support barrel; SCC, IASCC, corrosion, and neutron embrittlement for the core support barrel upper flange; SCC, IASCC, and corrosion for the core shroud; SCC, IASCC, corrosion, and stress relaxation for the core shroud assembly bolts; SCC, IASCC, and corrosion for the core shroud tie rods; SCC, IASCC, and corrosion for the fuel alignment plate guide lugs; SCC, IASCC, and corrosion for the core support plate; SCC, IASCC, corrosion, and stress relaxation for the fuel alignment pins; SCC, IASCC, and corrosion of the lower support structure beam assemblies; SCC, IASCC, and corrosion for the core support columns; SCC, IASCC, corrosion, neutron embrittlement, and stress relaxation of the core support column bolts.

10. Section 4.3.1.2 of the LRA indicates that a component level scoping and component pre-evaluation were not applied to the RVI before the aging evaluation to determine which components were subject to an AMR. Instead, all components of the RVI were initially included in the AMR. Section 4.3.1.2 of the LRA further indicates, "some components were determined not to be within the scope of license renewal since they are not required for the RVI to perform their intended function." Describe which components were considered to be outside the scope of license renewal and clarify the criteria that were used to conclude that these components were not required for the RVI "to perform their intended function." Identify the components that provide a structural integrity function.

11. Section 4.3.2 of the LRA indicates that IASCC is not plausible for Calvert Cliffs RVI because IASCC has not been observed for components with the temperature, oxygen and radiation levels present for the Calvert Cliffs RVI, either in operating plants or in

laboratory tests. Identify the operating plant experience and laboratory test data that forms the basis for this conclusion. Identify the RVI components at Calvert Cliffs that are subject to a neutron fluence greater than 5x1 E20 n/cm2. For these components, identify the temperature, oxygen, radiation levels and stress levels. What inspections or aging management programs (AMP) will be performed for these components during the extended period of operation to ensure that these components do not exhibit IASCC during the license renewal term?

How does the information in Information Notice 98-11, "CRACKING OF REACTOR VESSEL INTERNAL BAFFLE FORMER BOLTS IN FOREIGN PLANT" impact this evaluation? Since bolt cracking has occurred at the junction of bolt head and shank, which is not accessible for visual inspection, how will core shroud and bolts (CEASB) and other RVI bolting that is subject to IASCC be examined? What inspections or aging management programs (AMP) will be performed for these components during the extended period of operation to ensure that these components do not exhibit IASCC during the license renewal term?

12. Section 4.3.2 of the LRA indicates, "procedures will be enhanced if modified to specifically identify each component of the RVI which relies on this program for aging management for license renewal." Which RVI components have had or will have their procedures modified as a result of the review of aging management for license renewal? Briefly summarize the reasons for the modifications.

13. Section 4.3.2 of the LRA indicates that of the three U.S. suppliers of light water reactor the most fatigue-susceptible RVI components have been identified for pressurized water reactor (PWR) plants. What is the most-fatigue susceptible RVI component? Explain how this was determined? If the usage factor for these components exceeds 0.5 (criteria specified in the LRA), what additional actions will be initiated. Additionally, indicate to what degree would the scope of components being evaluated be expanded as a result of exceeding the usage factor of 0.5 for the components normally evaluated.

14. Section 4.3.2 of the LRA indicates, "Thermal aging is potentially significant for: (1) centrifugally-cast parts with delta ferrite content above 20%; (2) statically-cast parts with molybdenum content meeting CF3 and CF8 limits and with a delta ferrite content above 20%; and (3) statically-cast parts with molybdenum content exceeding CF3 and CF8 limits with delta ferrite content above 14%." Provide the basis for the conclusion that thermal aging is not significant below these levels. How is the amount of delta ferrite in cast stainless steel RVI components be determined? What are the uncertainties in these test methods? How are the uncertainties incorporated into the estimate of the delta ferrite?

If the delta ferrite values exceed the limits in the LRA, Section 4.3.2 indicates that an examination will be performed. Provide a fracture mechanics analysis to demonstrate the critical flaw size at the end of the license renewal term for these limits. Identify the inspection procedures and the capability of the inspection to detect flaws smaller in size than the critical flaw size.

15. Section 4.3.2 of the LRA indicates "A stress analysis will be performed specifically to evaluate the potential for SSC of CEA shroud bolts." Provide the criteria that will be used in this evaluation. Provide the data that will be used to establish the criteria that A-286 CEA shroud bolts are not subject to SCC during the extended period of operation. What type of examination, extent of examination and acceptance criteria are applicable for A-286 CEA shroud bolts under the ARDI program?

16. Table 4.3-2 indicates erosion, erosion/corrosion, general corrosion/uniform attack, hydrogen damage and pitting/crevice corrosion are not plausible. Explain the bases for these conclusions.

17. Section 4.3.2 indicates stress corrosion cracking/IGSCC/intergranular attack are potential age related degradation mechanism(s) (ARDM(s)) for RVI components fabricated from AMS 5735 iron base superalloy A-286; but does not identify any Inconel 600 components. Primary water stress corrosion cracking in PWR environment has occurred in Inconel 600 components. Identify the reactor vessel internal components that were fabricated using this material or other nickel base alloys and describe the aging management program that will be used during the extended period of operation to ensure these components are not susceptible to primary water stress corrosion cracking.

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.

19. Section 4.3.2 states that "No instances of degradation of RVI for PWRs have been recorded which have definitely been attributed to neutron embrittlement," and "Calvert Cliffs has not discovered any thermal-aging related damage for the RIV. Also there have not been RVI damage events at other PWRs that were identified as thermal aging failure." Based on the staff's experience the degradation in material properties attributable to these two ARDMs can only be "observed" through evaluation of the results of destructive material property testing of degraded components. Therefore, elaborate on the basis for these conclusions.

20. Section 4.3.1.1 of the LRA indicates that the aging evaluation of RVI "credits" the primary water chemistry control as an Aging Management Program to manage aging of RVI components. Which components and ARDMs are affected by primary water chemistry control? Describe the "credits" assumed for each ARDM and affected component and justify the credits assumed.

21. Section 4.3 indicates that changes in the design of the hold down rings (HDRs) installed at Calvert Cliffs Units 1 and 2 were made as a result of wear experienced in a similar component at another reactor plant and the discovery for the need to provide for additional fuel assembly growth. Table 4.3-1 identified the HDRs as a device type subject to AMR. Table 4.3-2 identifies the HDRs as device types subject to wear as an ARDM. Further, the LRA indicates that wear can be discovered when the reactor vessel is opened during a refueling outage, and the RVI are subject to a visual examination of accessible surfaces.

    The HDR is a near flat ring spring of a rectangular cross section. The hold down force is developed by deflecting the inner and outer edges of the ring spring in a direction because flattening of the ring. In deflecting the HDR, the outer edge of the top surface and inner edge of the bottom surface of the ring contact and load the Pressure Vessel Closure and the Upper Guide Structure (UGS) flange, respectively. Provide a description of the accessibility to the bottom surface of the HDR that contacts the UGS flange, the UGS flange and the undersurface of the vessel closure for visual inspection. Your description should account for the accuracy required in the use of visual indications of detectable wear to reliably determine changes in the HDR load developing capability.

    In addition, any such wear, if it occurs, may gradually reduce the HDR clamping force and induce core barrel motion under flow excitations. Verify the existence of a program for monitoring and trending the possible core barrel motion, using data from excore neutron detectors.

22. Provide the basis for considering the HDR as a device type subject to stress relaxation. Describe any inspections performed, or that will be performed with regard to changes in as-built dimensions or deflection measurements that demonstrate that the hold down load provided by the HDR has not and will not be reduced to impair its intended function during the period of extended operation.

23. Describe the visual examinations of the CEASB that have been previously performed or that will be performed to maintain the structural integrity of the RVI consistent with the current licensing basis during the period of extended operation. Describe the portions of the CEASB that are accessible for visual examination and discuss how the observations can be used to reliably demonstrate and provide adequate assurance that neutron embrittlement will be managed during the period of extended operation.

24. Are there any parts of the systems, structures and components within the RVI 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.

September 3, 1998

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

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'98 OCT -1 P3:1

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR THE SERVICE WATER SYSTEM (TAC NOS. M99591, M99592, AND M99210)

Dear Mr. Cruse:

By letter dated August 21, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Service Water System (5.17) 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 (NRC) staff review the Service Water System (5.17) 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 Service Water System (5.17) 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 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,

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

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Docket Nos. 50-317 and 50-318
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Exhibit 10

Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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DSolorio (DLS2)
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TMarsh (LBM)
GHubbard (GTH)
WLefave (WTL1)
Yeuh-Li Li (YCL)
KParczewski (KIP)

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

1. Section 5.17, indicates that a previously performed evaluation concluded that the non-safety-related portions of the service water system (SRW) are adequately rugged to withstand a design basis earthquake, which is credited in the design basis for preserving system inventory. The same section also indicates that all safety-related portions of the SRW are within the scope for license renewal. Since the non-safety-related portions of the SRW piping are credited in preserving system inventory during a design basis earthquake, it is not clear why this portion of piping is not within the scope for license renewal. Provide the basis for excluding this portion of SRW from the scope of license renewal or a cross reference to where it is addressed in the license renewal application (LRA).

2. Section 5.17.2, indicates that the safety-related SRW system piping will be included in an Aging-Related Degradation Inspection (ARDI) program to verify that degradation of the piping is not occurring, and the results of that inspection will be evaluated for applicability to the non-safety-related SRW piping. In addition, you state that the non-safety-related portions of SRW piping and the safety-related piping were both originally designed to USAS B31.1 and both are subject to the same environmental service conditions and chemistry controls. The applicability evaluation will also consider, at a minimum, flow rate and configuration differences between safety-related and non-safety-related SRW piping. Please clarify how the flow rate and configuration differences between safety-related and non-safety-related SRW piping will be considered in the applicability evaluation, and clarify the basis upon which you concluded that the results of the inspection of the safety-related piping are adequately representative of the aging degradation of the non-safety piping.

3. According to Subsection 5.17.1.1, the SRW piping to the instrument and plant air compressors and aftercoolers is within the scope of license renewal for fire protection. However, a failure anywhere in the SRW supply or return piping to these components (or any connected systems or components) can affect not only the fire protection safe shutdown, but also all other safe shutdown events requiring the operation of the SRW system. Clarify the basis for determining why the SRW system piping to the compressors and aftercoolers is within the scope of license renewal for fire protection, but not within the scope for the SRW.

4. In Section 5.17.1.3, you have identified that the only passive function associated with the SRW system not otherwise dispositioned is "to maintain the pressure boundary of the system liquid." In light of your response to Component Cooling Water System RAI No. 2 (letter dated August 1, 1998), and the air-operated components in the SRW system,

identify how the aging management review has been conducted for the air-operated components in the SRW system.

5. Are there any parts of the SRW systems, structures or components 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: (1) preventive actions that will mitigate or prevent aging degradation; (2) parameters monitored or inspected relative to degradation of specific structure and component intended functions; (3) detection of aging effects before loss of structure and component intended functions; (4) monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) acceptance criteria to ensure structure and component intended functions; and (6) operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

6. Section 5.17, indicates that the SRW system was designed to USAS B31.1 Code requirements. While B31.1 does not require an explicit fatigue analysis, it does specify allowable stress levels based on the number of anticipated thermal cycles. Table 5.17-3 indicates that fatigue is not a plausible age-related degradation mechanism (ARDM) for the SRW system. Because fatigue is normally treated as a Time-Limited Aging Analyses in accordance with the requirements of 10 CFR 54.21(C), please provide the basis for concluding fatigue is not a plausible ARDM for SRW components.

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.

September 4, 1998

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

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'98 OCT -1 P3:23

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR FIRE PROTECTION SYSTEM (TAC NOS. MA1445, MA1446, AND M99214)

Dear Mr. Cruse:

By letter dated March 27, 1998, Baltimore Gas and Electric Company (BGE) submitted for review the Fire Protection System (5.10) 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 (NRC) staff review the Fire Protection System (5.10) 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 Fire Protection System (5.10) 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 related to scoping 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
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Mr. Charles H. Cruse
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Unit Nos. 1 and 2

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TCheng (TMC)
SLittle (SLL)

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

1. Section 5.10, "Fire Protection," addresses the fire protection (FP) functions and the safe shutdown function (Appendix R to 10 CFR 50). Describe how the Calvert Cliffs Fire Protection Plan, which is required under 10 CFR 50.48, "Fire Protection," was used in developing the system-level scoping and the integrated plant assessment (including FP and safe shutdown).

2. 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.

3. Identify the fire protection components, if any, that have been excluded from the scope of the rule because they are subject to replacement based on qualified life or a specified time period as permitted under 10 CFR 54.21(a)(1)(ii).

4. Describe, in detail, how the post-fire remote or auxiliary shutdown panels were addressed in the system level scoping process and the aging management review process.

Enclosure

September 4, 1998

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 & 2, INTEGRATED PLANT ASSESSMENT FOR HEATING AND VENTILATION SYSTEMS (TAC NOS. MA1018, MA1019, MA1034, MA1106, MA1107, M99224, MA1040, MA1041, AND MA1035)

Dear Mr. Cruse:

By letter dated April 8, 1998, Baltimore Gas and Electric (BGE) submitted for review its license renewal application. The staff has reviewed Section 5.11A, "Auxiliary Building Heating and Ventilation System;" Section 5.11B, "Primary Containment Heating and Ventilation System;" and Section 5.11C, "Control Room and Diesel Generator Buildings HVAC," of Appendix A to the application against the requirements of 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(3). 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

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SDroggitis (SCD)
DSolorio (DLS2)
PDLR Staff

TCheng (TMC)
EHylton (EGH)

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS UNITS 1 AND 2 INTEGRATED PLANT ASSESSMENT
SECTION 5.11A, "AUXILIARY BUILDING HEATING AND VENTILATION SYSTEM,"
SECTION 5.11B, "PRIMARY CONTAINMENT HEATING AND VENTILATION SYSTEM," AND
SECTION 5.11C, "CONTROL ROOM AND DIESEL GENERATOR BUILDING HVAC"
DOCKET NOS. 50-317 AND 50-318

1. Sections 5.11A.1, 5.11B.1 and 5.11C.1 of the application state that representative historical operating experience pertinent to aging is included in appropriate areas, to provide insight supporting the aging management demonstration. From the past operating experience, provide specific examples of how the corrective actions (including types, methods, criteria, etc.) related to the aging degradation of heating and ventilation (H&V) systems were taken in the auxiliary building, primary containment, and control room and diesel generator buildings.

2. As described in Sections 5.11A and 5.11C of the application, for the H&V systems located in the auxiliary building, and control room and diesel generator buildings, some cracking has been discovered in plant heating, ventilating, and air conditioning (HVAC) ducting due to vibration-induced fatigue. The application also states that these isolated failures were due to a combination of design and installation deficiencies. Please address the following:

   a. Clarify the basis for the conclusion that these isolated failures did not involve any age-related degradation mechanisms (ARDMs).

   b. With regard to the corrective actions, provide details of how these cracks were corrected and how these failures affected the intended function.

3. As described in the operating experience for Sections 5.11A, 5.11B, and 5.11C of the application, loosening of fasteners due to dynamic loading was identified as an ARDM. Provide a justification of why this ARDM is identified as plausible only for fans in the ARDMs tables (Tables 5.11A-2, 5.11B-2 and 5.11C-2) and not fasteners or other device types exposed to dynamic loads.

4. Sections 5.11A.2, 5.11B.2 and 5.11C.2 of the application describe ARDM and device type combinations for aging management. Provide a justification as to why mechanical wear of the duct systems is not considered as a plausible ARDM.

5. As described in the application (Sections 5.11A.1.3, 5.11B.1.3 and 5.11C.1.3), some of the device types (such as damper, filters, hand valve, and pressure differential indicator in the auxiliary building; damper, filter and solenoid valve in the primary containment; analyzer element, gravity damper, hand valve and temperature transmitter in the control room and diesel generator buildings) are subject to a detailed evaluation of ARDMs as part of the aging management review (AMR). However, there are no entries of potential and plausible ARDMs under these device types in Tables 5.11A-2, 5.11B-2 and 5.11C-2

of the application. Provide a summary description of the ARDMs considered for these device types and the basis for the plausible ARDM conclusion.

6. In describing the aging management programs for components such as ducting and heat exchangers, the application (Discovery in Pages 5.11A-13, 5.11B-15 and 5.11C-11) states that crevice corrosion, general corrosion, and pitting can be readily detected through visual examination. Clarify how these aging effects will be managed for locations such as lap joints that cannot be readily inspected visually.

7. Tables 5.11A-1, 5.11B-1 and 5.11C-1 of the application list all the H&V system device types for which the AMR is required. Also, Sections 5.11A.1.3, 5.11B.1.3 and 5.11C.1.3 of the application include a statement that only the pressure-retaining function (the passive intended function) for these device types is considered in the AMR for the H&V systems in the auxiliary building, primary containment, and control room and diesel generator buildings. However, no description of how to maintain this passive intended function is included in the application. Clarify how the aging management programs described in the application maintain the pressure-retaining function of these device types.

8. Pages 5.11A-7 and 5.11C-6 of the application indicate that certain device types "do not require a detailed evaluation of specific aging mechanisms because they are considered part of a complex assembly whose only passive function is closely linked to active performance." The listed device types include accumulators, piping, and valves. Clarify how the passive functions of these devices are adequately managed by such performance monitoring. In particular, describe the nature of the monitoring and demonstrate that the degradation of the particular component intended function is "closely linked" to the parameters being monitoring in a performance monitoring program, such that the component intended function would be maintained for the period of extended operation.

9. Page 5.11B-6 of the application indicates that temperature elements do not require an AMR because they have only active functions. However, thermocouples and RTDs are installed in thermowells which perform a pressure-retaining function and have housings which serve as environmental barriers. Clarify BGE's basis for concluding that temperature elements do not have any passive functions.

10. On Page 5.11B-10, the application includes a description of two aging degradation experiences for valves: (1) some wear of the containment purge supply and exhaust containment isolation valves (control valves) were identified, and (2) check valves have experienced pressure boundary failures with several valves failing back-leakage tests. The application also states that the root cause of these failures is due to a combination of wear and misapplication of the valve for its intended function. Please address the following:

    a. Clarify the basis for the conclusion that these failures did not involve any age-related degradation mechanisms.

b. Provide a description of the corrective actions implemented for these two cases.

11. Are there any parts of the systems, structures and components within the H&V systems 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.

September 4, 1998

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

PUBLIC DOCUMENT ROOM

'98 SEP 28 P2:11

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT ON GENERIC SAFETY ISSUES (TAC NO. MA2156)

Dear Mr. Cruse:

By letter dated April 8, 1998, Baltimore Gas and Electric Company (BGE) submitted for review its license renewal application. Based on a review of the information submitted, the Nuclear Regulatory Commission (NRC) staff has identified in the enclosure, areas regarding unresolved generic safety issues 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 NRC 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,

/S/

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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Exhibit 13

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DSolorio (DLS2)
PDLR Staff

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

1. Describe the BGE process and criteria for determining which unresolved generic safety issues (GSIs) listed in NUREG-0933, "A Prioritization of Generic Safety Issues," should be reviewed to identify any concerns that may be related to the effects of aging or time-limited aging analyses for systems, structures or components within the scope of license renewal.

2. Discuss whether BGE specifically evaluated GSI-23, "Reactor Coolant Pump Seal Failures," and GSI-173.A, "Spent Fuel Storage Pool: Operating Facilities," as relating to the license renewal aging management review or time-limited aging evaluation, as described in an NRC staff letter to the Nuclear Energy Institute (NEI), dated January 29, 1998. If yes, identify where these GSIs are evaluated in the application or describe the BGE evaluation results. If not, provide the justification that such an evaluation is not warranted.

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.

Enclosure

September 7, 1998

Mr. Charles H. Cruse, Vice President
Nuclear Energy Division
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1650 Calvert Cliffs Parkway
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'98 OCT -5

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 & 2, COMMODITY REPORTS FOR COMPONENT SUPPORTS AND PIPING SEGMENTS THAT PROVIDE STRUCTURAL SUPPORT (TAC NOS. MA0291, MA0292, AND M99204)

Dear Mr. Cruse:

By letters dated October 22, 1997, and March 27, 1998, Baltimore Gas and Electric Company (BGE) submitted for review the Component Supports (3.1) and Piping Segements that Provide Strucutral Support (3.1A) commodity reports, respectively, as attached to the "Request for Review and Approval of System and Commodity Reports for License Renewal." BGE requested that the Nuclear Regulatory Commission (NRC) staff review reports 3.1 and 3.1A to determine if the reports meet 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 reports 3.1 and 3.1A against the requirements of 10 CFR 54.21(a)(1), and 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
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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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Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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GBagchi (GXB1)
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MZobler/RWeisman (MLZ/RMW)
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RArchitzel (REA)
CCraig (CMC1)
LSpessard (RLS)
RCorreia (RPC)
RLatta (RML1)
EHackett (EMH1)
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SDroggitis (SCD)
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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 3.1.1 - Scoping

1. Table 3.1-1 contains a list of systems within the scope of license renewal that contain component supports within the commodity evaluation cover under Section 3.1 of the license renewal application (LRA). This list was compared to the list of all the systems within the scope of license renewal. This review revealed five system (System 68, Spent Fuel Storage; System 70, Refuel Pool; System 76, Secondary Sampling System; System 103, Emergency Diesel Generator HVAC; and System 120, Barriers and Barrier Penetrations) that were identified as being within the scope of license renewal but not having component supports within the scope of the commodity evaluation provided in Section 3.1 of the LRA. Please identify the scope of component supports from these five systems that are included within the scope of the aging management review under the component supports commodity groups. Indicate whether any of these five systems have no component supports that require an aging management review.

Section 3.1A - Scoping

2. Subsection 3.1A.1.1 includes a statement that the system's seismic structural boundary extends beyond the valve to the first seismic anchor or "equivalent." Provide a discussion to explain what kind of piping support arrangement is "equivalent" to the "seismic anchor."

Section 3.1.2 - Aging Management Review

3. Page 3.1-1 of Section 3.1 described that the Seismic Qualification Utility Group (SQUG) guidance was used as one of the sources for grouping the component supports and was used for the baseline inspections. As stated on Page 5 of supplemental safety evaluation report No. 2 (SSER-2) on SQUG's Generic Implementation Procedure, Revision 2 (GIP-2) dated May 22, 1992, the qualification of seismic adequacy of equipment (including supports) in older operating nuclear plants does not address the aging effects of equipment. The SSER-2 also stated that the staff will not accept any claim that the experience data collected by the SQUG for the Unresolved Safety Issue (USI) A-46 program adequately addressed the aging effects of equipment. Provide a justification for using the GIP-2 for scoping component supports.

4. Page 3.1-1 of subsection 3.1.1.1 provides the definition of component supports, "a component support is defined as the connection between a system, or component within a system, and a plant structural member (e.g., the concrete floor or wall, structural beam or

column, or ground outside the plant buildings)." From the review of Section 3.1, it is not clear that the steel structural frames used for the support of piping systems were treated as component support or as structural components. If the steel structural frames are considered as components, which of the aging management programs (existing or new) will be used for managing the aging effects of the steel structural frames.

5. Page 3.1-1 in Section 3.1 includes a statement that the structures aging management review (AMR) considered the effects of aging caused by the surrounding environment, while the component supports AMR considered the effects of aging caused by the supported equipment (thermal expansion, rotating equipment, etc.) as well as the surrounding environment. Clarify how the aging effects of the supported equipment was considered in the AMR for the component supports.

6. Please address the following questions related to the commodity description and the boundary for component supports:

   a. Are all types of fasteners (such as bolts, nuts, clips, clamps, brackets, etc.) used to attach component supports to components and component supports to structures included within the scope of component supports requiring an AMR? If so, in what section of the LRA is the AMR of fasteners addressed? If not, provide a justification for not including fasteners within the scope of an AMR.

   b. For fasteners that rely on welded connections to the components (e.g., pipe stanchions with welded attachment to a pipe or a piece of equipment), identify if the welds are considered part of the fastener, component being supported or the component support. Identify where in the LRA is the AMR for these fasteners and welds are located, or provide a justification for not including these fasteners and welds within the scope of an AMR.

   c. Structural steel members such as supplementary steel members (e.g., heating, ventilation, and air conditioning (HVAC) duct supports labeled as "rod hanger trapeze supports) are not identified as within the scope of component supports. Identify where in the LRA the AMR for these components is addressed or provide a justification for not performing an AMR of these components.

7. Page 3.1-2 in subsection 3.1.1.1 includes the statement that supports for tubing are included in Section 6.4 of the LRA entitled "Instrument Lines." How is the distinction made (or boundary) between piping and tubing for defining the scope to be covered under Section 3.1 versus Section 6.4?

8. Table 3.1-1 on page 3.1-3 defines the systems within the scope of license renewal containing supports within the commodity evaluation. This table does not include the steam generator blowdown system; containment isolation group; control room and diesel generator building HVAC systems. Identify the section within the LRA that addresses the AMR for the associated supports for these systems and structures or provide a justification for not including them within the scope of components requiring an AMR.

9. In Table 3.1-2, only the rod hanger trapeze supports are listed for the HVAC ducting supports. Based on the staff's experience, unistrut type of supports are widely used for the HVAC ductworks in operating nuclear power plants. Clarify if the rod hanger trapeze type of support is the only type of support used for the HVAC ducting systems. If any other type of support is used for the HVAC ducting systems, identify where in the LRA these supports are addressed or provided a justification for not subjecting these components to an AMR.

10. Table 3.1-2 only identifies ring foundations for supports of the flat-bottom field-erected vertical tanks. Provide the basis for not considering the degradation due to aging (loose anchors, general corrosion of anchor chairs and long anchor bolts, etc.) of the anchorage systems in the component support AMR.

11. Provide a discussion of how dynamic loading (e.g., vibrations) aging effects for the anchorage systems of elements inside electrical cabinets (such as relays) are managed.

12. Table 3.1-3 indicates that general corrosion is not plausible for frames and saddles. Please provide the basis for this conclusion.

13. Based on the staff's experience, "loose bolts" (high strength bolts, anchor bolts, etc.) due to vibration is a common type of aging effect of component supports with bolt-connections. Provide the basis for excluding this as an applicable aging effect. If applicable, please include a discussion of how the plant operating and maintenance history support this conclusion.

14. Provide the basis for excluding concrete cracking as an applicable aging effect requiring an aging management program for the flat-bottom vertical tank ring foundation.

15. Regarding expansion anchors and embedded anchors, which are commonly used for the connections between the component supports and structural components (walls, floors and beams), please clarify the following:

    a. Any loss of clamping force over time (age-related degradation) associated with expansion anchors should be properly managed, because it will affect the stiffness properties of supports and will change the behavior of components under dynamic loading such as an earthquake. Please clarify how the loss of clamping force was addressed in the AMR for these components. If not addressed, provide the basis for not addressing the loss of clamping force for these components.

    b. The cracking of surrounding concrete (age-related degradation) that typically occurs around concrete expansion and embedded anchors was not identified as a potential aging effect. Provide the basis for not considering this as a potential aging effect.

c. Provide the basis for not including corrosion of steel chairs, loose long anchor bolts, and deterioration of the nozzle between tanks and connected pipes within the scope of an AMR. Based on the staff's experience, these components would have been expected to be addressed within the "support/ARDM combination" Group 6.

16. Please clarify the following questions related to the baseline walkdowns or inspections described on pages 3.1-6 and 3.1-7.

a. Will the baseline walkdowns (or inspections) involve any actions other than visual observations? If not, explain how will cracks associated with incipient fatigue failures or with bolt cracking be detected.

b. What parameters will be reviewed and/or inspected and what acceptance criteria will be used?

c. Please clarify how the baseline procedure implements expansion of the sample size and scope based on the findings from the initial sampling? For example, if an age related degradation mechanism (ARDM) is identified for a specific support-type sample, then will all supports for that "type" be inspected and will the scope be increased for other support types having a similar environment, design, or loading?

d. General corrosion of steel is identified as an ARDM that applies to all support types. Will every support be included in the baseline walkdown/inspection? If not, describe the process and the basis that will be used to determine the walkdown/inspection sample size?

e. The LRA states that follow-on will be undertaken if evidence of significant aging is found. Clarify what is meant by significant aging? Provide examples of "significant aging" and what elements would be included in the follow-on actions. Also, what actions are taken if the identified ARDM is not significant at the time of baseline inspection?

17. Table 3.1-3 lists potential and plausible ARDMs for component supports. Please clarify the following related to the headings and potential ARDMs.

a. Provide the basis for excluding mechanical wear as an ARDM for supports containing pins, springs, sliding plates, etc., from an AMR.

b. For general corrosion of steel, is corrosion attack by any medium other than water or moisture considered (e.g., chemical attack due to leaks, spills, or effluents)?

c. Are any materials other than steel and elastomer elements used in component supports (e.g., Teflon coated or Lubrite plates)?

d. Did you include thermal striping and thermal stratification in your assessment of thermal expansion loading? If not, provide a justification for excluding these ARDMs from the scope of your AMR.

e. Referring to the ARDMs shown in Table 3.1-3, discuss the consideration given to possible interaction between individual ARDMs. (As an example, vibratory loads in conjunction with irradiation embrittlement might be a very critical combination.)

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.

19. Based on the staff's experience, a large number of frame types of piping supports are fabricated with threaded fasteners. If the bolted piping frame supports are used, clarify the basis for the following conclusion: "the aging effects are not expected to prevent the piping frames from performing their intended support function," described on page 3.1-11; and the conclusion, "while hydraulic vibration or water hammer and thermal expansion have been observed, the aging effects are not expected to prevent the pipe frames from performing their intended support function and these ARDMs are considered to be not plausible for this kind of supports," on page 3.1-12.

20. A statement was made in the application that the American Society of Mechanical Engineers (ASME) Section XI inservice inspection for component supports includes a visual examination of a prescribed sampling of the systems covered by the program. In addition to the sampling criteria adopted from the ASME Code, Section XI (as stated in the last paragraph of Page 3.1-14), provide a description of the criteria for sample expansion

(how the sample size of component supports are to be expanded when degradations are identified discussed on Page 3.1-15)

21. With regard to the description in the second paragraph of Page 3.1-17, please clarify the following:

    a. The second sentence states that the sample approach will be comparable to the approach required by ASME Section XI for piping supports of ASME Class 3 systems. Clarify the definition of the word "Comparable." Identify the specific differences and describe why are these differences being implemented.

    b. The fourth and fifth sentence state that these walkdowns document the condition of the piping supports within the scope of license renewal for all piping support types, except piping frames outside the containment. If an active corrosion mechanism is found during the additional sampling baseline walkdowns for pipe hangers outside the containment, then the inspection scope for that system would be expanded to pipe frame supports outside containment. Provide justification why the pipe frame supports outside the containment are included in the scope only when an active corrosion mechanism is found during the additional sampling baseline walkdowns for pipe hangers outside the containment.

22. Page 3.1-18 states that "None of the failure modes is expected to be affected by age-related effects, such as anchor-bolt relaxation or concrete shrinkage because:" bolt preload in the anchor is not counted on for anchor function. Once an anchor is "set" by torque, anchor function is maintained by the irreversible expansion of the anchor expansion ring or cone into the concrete. Summarize the information that provides the basis for this conclusion. Based on the staff's experience, once the anchor bolt is "set," the result of anchor-bolt relaxation or concrete shrinkage will cause the anchor-bolt function change due to the reduction of anchor-bolt stiffness (the stiffness of anchor-bolt systems will decrease with time, and only the anchor strength is maintained.) and, in turn, the reduction of anchor-bolt stiffness will modify the dynamic behavior of the supported components.

23. Page 3.1-23 states that the Group 2 "support/ARDM combination" includes all 15 component support types within the three component support groups (cable-tray supports, HVAC ducting supports, and equipment supports). This section also provides a description on page 3.1-26 that the aging management approach for the three component support groups rely on inspections performed by the seismic verification program (SVP) for eight support-types, inspections performed by the inservice inspection (ISI) program for three support-types, and additional sampling baseline walkdowns for two support-types. Please clarify what are the two support types that are not covered by these three baseline walkdown activities. Are they the two support types for which no baseline walkdowns were required? If so, what is the basis for this determination?

24. Normally, resistance or susceptibility to stress corrosion cracking of high strength steel bolts is established by hardness of the bolt material. Discuss what plans, if any, do you have to check the hardness of the bolts either from in-place bolts or bolting in the

warehouse. If applicable, provide hardness data for the bolting material as necessary to support the response.

25. Describe the visual inspection activities performed during the SVP walkdown that were used to identify potential ARDM effects such as loosening of bolted connections or loss of weld integrity. Please identify what documentation is used to implement these inspection activities.

26. Section 3.1.A.2 indicates that piping segment beyond the safety-related/nonsafety-related (SR/NSR) boundary to the first seismic restraint is considered as structural support for the system pressure boundary isolation valve. Therefore, piping segments beyond the SR/NSR boundary are classified as Seismic Category I up to and including the first seismic anchor. This section further states that given the similarity of the piping materials for piping within the SR pressure boundary, to these outside this boundary that are designed and maintained to SR requirements, any material degradation identified on the piping segments within the SR pressure boundary would lead to an evaluation for generic implications on the NSR side of this boundary. The staff interprets this statement as a commitment that the licensee will evaluate these NSR piping segments only if some aging degradation has been identified on the SR piping segments. Since these NSR piping segments have the intended safety function of providing structural support to the SR piping and boundary isolation valves, provide a justification for not performing the applicable aging management activities for detecting applicable aging effects of the NSR piping independent of degradation identified with the SR piping.

27. Section 3.1.2 states that some supports are inaccessible either because they are located underwater (in spent fuel pools or refueling water storage tanks), or because they are located in high radiation areas. The section further states that it may not be possible to perform a visual walkdown of these supports. However, other inspection techniques (e.g., remote video) may be recommended under the age-related degradation inspection (ARDI) program if they are viable. The ARDI program will either sample some of these supports, sample other accessible supports that are similar in design and/or environment, or will provide an analysis that will document why any inspection is not required. Please summarize the scope of the inspection activities, the inspection methods to be used, the frequency of inspections, the criteria used to determine that frequency, and the basis for this criteria related to the visual inspections/walkdowns activities. If an analysis has been used to determine that an inspection is not needed, provide sufficient information related to the analysis to justify this determination.

28. Are there any parts of the systems, structures, or components described in this section 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: (1) Preventive

actions that will mitigate or prevent aging degradation; (2) Parameters monitored or inspected relative to degradation of specific structure and component intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

September 7, 1998

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

SUBJECT: CLARIFICATION REGARDING SELECTED FEEDWATER AND DIESEL FUEL OIL REQUESTS FOR ADDITIONAL INFORMATION RESULTING FROM MAY 6, 1998, MEETING WITH BALTIMORE GAS AND ELECTRIC COMPANY (TAC NOS. M95453, M95454, M99178, M95457, M95458, AND M99180)

Dear Mr. Cruse:

On May 6, 1998, several members of the Nuclear Regulatory (NRC) staff and Baltimore Gas and Electric Company (BGE) staff meet to discuss BGE's proposed responses to the NRC staff's requests for additional information (RAI) issued on the BGE integrated plant assessment technical reports Feedwater System (5.9) and Diesel Fuel Oil System (5.7) integrated plant assessment technical reports submitted by letters dated February 13, 1998, and February 19, 1998, respectively.

As a result of the meeting, the NRC staff agreed to provide additional clarification on these two RAI. This letter provides the staff's additional clarification and incorporates new information gained by the staff through the review of BGE's license renewal application submitted by letter date April 8, 1998.

To facilitate tracking of the Feedwater System (5.9) and Diesel Fuel Oil System (5.7) RAI responses, the item numbers of revised questions correspond to the item numbers of the RAI submitted to BGE by letters dated February 13, 1998, and February 19, 1998, respectively.

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

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Mr. Charles H. Cruse
Baltimore Gas & Electric Company

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SDroggitis (SCD)
RArchitzel (REA)
CCraig (CMC1)
LSpessard (RLS)
RCorreia (RPC)
RLatta (RML1)
EHackett (EMH1)
AMurphy (AJM1)
TMartin (TOM2)
DMartin (DAM3)
GMeyer (GWM)
WMcDowell (WDM)
SStewart (JSS1)
THiltz (TGH)
SDroggitis (SCD)
DSolorio (DLS2)
PDLR Staff

WLefave (WTL1)
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REQUESTS FOR ADDITIONAL INFORMATION

CLARIFICATION AND FOLLOWUP
REGARDING
BALTIMORE GAS AND ELECTRIC COMPANY
LICENSE RENEWAL APPLICATION

Section 5.9, Feedwater System

The following items are clarifications and/or revisions to requests for additional information (RAI) submitted to Baltimore Gas and Electric Company (BGE) by letter dated February 13, 1998 for the integrated plant assessment report, 5.9, Feedwater System (FWS). BGE submitted the FWS integrated plant assessment to the Nuclear Regulatory Commission (NRC) by letter dated May 23, 1997.

1. Bulletin 79-13 discusses stress assisted corrosion in pressurized water reactor FWS, and Generic Safety Issue 14 discusses stress corrosion cracking specifically in Combustion Engineering Plants Feedwater Systems. Provide a justification for not including stress corrosion cracking as an applicable aging effect for the FWS.

5. Clarify how the high level trip safety function is addressed in your application by identifying the components that have an intended function that supports a high level trip and identify where the aging management review is documented in the license renewal application (LRA).

8. This question is being withdrawn because it will be addressed by the applicant's proposed response to question number 4 given at the May 6, 1998, meeting.

9. Because the issue associated with fuses has generic applicability to license renewal, this question is being withdrawn until a final NRC staff position on fuses is developed. In the event that this staff position result in any matters affecting the BGE LRA, additional information will be requested at that time.

10. As a result of the May 6, 1998 meeting, the NRC recognizes that BGE's intent is to use device types in a similar manner as commodity groups; therefore, BGE is no longer requested to respond to this RAI. If specific concerns relating to the use of device types are identified, additional clarification will be requested at that time.

15. In order to ensure efficiency with respect to this question, the NRC staff is withdrawing this question until the detailed evaluation of the BGE's Chemistry Control program, as described in the LRA, is completed. As a result, this question will be reconsidered in light of the total informational needs relating to the staff's review of BGE's Chemistry Control program; any additional information on the Chemistry Control program will be requested at that time.

Enclosure

18. Describe the features of the wet and dry lay up process that ensures that the resulting conditions do not result in aging concerns. Consider in your response that secondary chemistry controls are not in place during wet and dry lay up and the potential effects on aging management.

19. The NRC staff recognizes that BGE acknowledged that corrosion is an applicable aging mechanism for carbon steel fasteners due to the exposure of these fasteners to the internal environment of borated systems. Discuss the potential for carbon steel fasteners being exposed to the internal environment of other plant systems such as the FWS. Based on the potential for exposure to the FWS internal environment, provide a bases for concluding that any applicable carbon steel and/or low alloy steel bolting within the scope of the FWS aging management review will not experience aging.

21. In order to ensure efficiency with respect to this question, the NRC staff is withdrawing this question until the detailed evaluation of the BGE's Chemistry Control program, as described in the LRA, is completed. As a result, this question will be reconsidered in light of the total informational needs relating to the staff's review of BGE's Chemistry Control Program; any additional information on the Chemistry Control program will be requested at that time.

24. In order to ensure efficiency with respect to this question, the NRC staff is withdrawing this question until the detailed evaluation of the BGE's Chemistry Control program, as described in the LRA, is completed. As a result, this question will be reconsidered in light of the total informational needs relating to the staff's review of BGE's Chemistry Control program; any additional information on the Chemistry Control program will be requested at that time.

32. In order to ensure efficiency with respect to this question, the NRC staff is withdrawing this RAI until the detailed evaluation of the BGE's Fatigue program, as described in the LRA, is completed. As a result, this question will be reconsidered in light of the total informational needs relating to the staff's review of BGE's Fatigue program; any additional information on the Fatigue program will be requested at that time.

33. In order to ensure efficiency with respect to this question, the NRC staff is withdrawing this RAI until the detailed evaluation of the BGE's Fatigue program, as described in the LRA, is completed. As a result, this question will be reconsidered in light of the total informational needs relating to the staff's review of BGE's Fatigue program; any additional information requests on the Fatigue program will be requested at that time.

34. In order to ensure efficiency with respect to this question, the NRC staff is withdrawing this question until the detailed evaluation of the BGE's Chemistry Control program, as described in the LRA, is completed. As a result, this question will be reconsidered in light of the total informational needs relating to the staff's review of BGE's Chemistry Control Program; any additional information on the Chemistry Control program will be requested at that time.

35. Discuss the significance of dropping below the minimum wall thickness criteria for the FWS with respect to the effectiveness of your aging management program. Provide a summary description of FWS-specific operating experience relating to occurrences of dropping below the minimum wall thickness criteria and a summary description of any corrective actions taken in response to these occurrences.

39. In order to ensure efficiency with respect to this question the NRC staff is withdrawing this question until the detailed evaluation of the BGE's Erosion Corrosion program, as described in the LRA, is completed. As a result, this question will be reconsidered in light of the total informational needs relating to the staff's review of BGE's Erosion Corrosion program; any additional information on the Erosion Corrosion program will be requested at that time.

Section 5.7, Diesel Fuel Oil System

The following are clarifications and/or revisions to requests for additional information (RAI) submitted to Baltimore Gas and Electric Company (BGE) by letter dated February 19, 1998 for the integrated plant assessment report, 5.7, Diesel Fuel Oil (DFO) System. BGE submitted the DFO system integrated plant assessment to the Nuclear Regulatory Commission (NRC) by letter dated May 23, 1997.

1. As a result of subsequent staff review of the DFO system report in light of the May 6, 1998 meeting, the following clarifications or additional information is requested to be submitted along with your response to this question. For the DFO system provide a summary description of the piping material, piping design standard, seismic category, pipe sizes, operating temperature and pressure, any leak detection measures, such as from inservice inspection and pressure tests, and any evidence of ground surface settlements adjacent to DFO piping.

4. As a result of subsequent staff review of the DFO system report in light of the May 6, 1998 meeting, the part of this question related to operating experience was revised as follows. The staff's review found that the inspected minimum bottom plate thickness for the fuel oil storage tank was found to be 0.247 inch greater than the required minimum thickness of 0.24 inch'. This measurement was taken after 20 years of service. How does this measurement compare with the baseline measurements or dimensions? Based on the current wear rates, provide a projection of the plate thickness after another 40 years of plant operation.

50-317

September 7, 1998

Mr. Charles H. Cruse, Vice President
Nuclear Energy Division
Baltimore Gas & 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, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT SECTIONS 3.3A, 3.3B, 3.3C, 3.3D, 3.3E, AND 6.2 (TAC NOS. MA1448, MA1449, MA1455, MA1089, MA1090, MA1098, MA1091, MA1092, MA1099, MA1094, MA1095, MA1100, MA1450, MA1452, MA1456, MA1443, MA1444, AND M99217)

Dear Mr. Cruse:

By letter dated April 8, 1998, Baltimore Gas and Electric Company (BGE) submitted its license renewal application. The NRC staff has reviewed reports 3.3A, "Primary Containment Structures," 3.3B, "Turbine Building Structure," 3.3C, "Intake Structure," 3.3D, "Miscellaneous Tank and Valve Enclosures" 3.3E, and 6.2, "Electrical Commodities," against the requirements of 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(3). By letter dated April 4, 1996, the Nuclear Regulatory Commission (NRC) staff approved Baltimore Gas and Electric Company'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,

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
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Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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LDoerflein (LTD)
BBores (RJB)
SDroggitis (SCD)
RArchitzel (REA)
CCraig (CMC1)
LSpessard (RLS)
RCorreia (RPC)
RLatta (RML1)
EHackett (EMH1)
AMurphy (AJM1)
TMartin (TOM2)
DMartin (DAM3)
GMeyer (GWM)
WMcDowell (WDM)
SStewart (JSS1)
THiltz (TGH)
SDroggitis (SCD)
DSolorio (DLS2)
PDLR Staff

TMarsh/GHubbard (LBM/GTH)
WLefave (WTL1)
GGeorgiev (GBG)
TSullivan (EJS)
KParczewski(KIP)
RWessman (RHW)
Yeuh-Li Li (YCL)
SLittle (SSL)

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

1. To facilitate the staff's review, BGE should provide a summary (in a tabular form) indicating which program (or programs) will cover safety-related tanks (including the field erected vertical tanks) and heat exchangers, and how these programs will be implemented.

2. For concrete components of Category I structures, the significant ARDMs are the following: settlement, freeze-thaw, leaching of calcium hydroxide, aggressive chemical attack, aggregate reaction, flowing water, and corrosion of embedded steel/rebar. The application addresses settlement as the applicable aging effect for concrete components of Category I structures only. Provide a brief summary (including basis and past operating experience, if any) to which these aging effects were either determined to be non-plausible or were not addressed, for the components described in sections 3.3A, 3.3B, 3.3C, and 3.3D.

3. Have the results of prior inspections (1994 and earlier) indicated any particular trend in the incidence of coating degradation or corrosion of steel?

4. One of the generic structural functions considered under the component level scoping process is to "Provide flood protection barrier (internal flooding events)." Is protection from external flood events an intended function? If so, clearly identify the structures associated with implementing the intended function and describe the corresponding aging management programs.

5. A heavy waterproofing membrane is provided at exterior walls and base slab. Rubber water stops are also provided at all construction joints up to grade elevation. Explain whether the waterproofing membrane and rubber water stops are relied upon to protect the concrete foundations. If not, provide the basis for excluding them from the scope of license renewal.

6. Subsurface drains are typically relied upon to lower the elevation of groundwater around the plant. Describe whether or to what extent the drainage system was considered to be within the scope of license renewal and if not, justify why. Summarize the operating experience of the drainage system and groundwater levels. Describe the consequence of elevated groundwater levels on the aging degradation of the various structures. Also,

Enclosure

provide a discussion of how failure of the drain system would impact the aging effects (such as settlements) that is considered not plausible.

7. As stated in the application, the need for a new aging management program for caulking and sealants which do not function as fire barriers was identified. Provide a description, in summary form, of this new program including the schedule for implementation, experience of failures of caulking and/or sealants, if any, that resulted in aging degradation of concrete and/or steel components, and corrective actions.

8. The modified aging management program Calvert Cliffs Nuclear Power Plant (CCNPP) Administrative Procedure MN-1-319, "Structure and System walkdowns," was credited (or will be credited) as an aging management program for seismic Category I structures. It is the staff's understanding that there are many safety-related reinforced concrete walls (e.g., auxiliary building walls, intake structure walls, etc.) in CCNPP. Provide the basis for why these safety-related reinforced concrete walls are not covered in the structure walkdown. (refer to Attachment 4 to MN-1-319, Structure Monitoring walkdown; Concrete Structures Other Than Containment [concrete slabs, beams, columns, base plates, and foundations])

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.

10. Section 3.3C.2 states that a structure performance assessment is currently required for Category I structures at CCNPP at least once every 6 years. Regulatory Guide (RG) 1.127 recommends a frequency of 5 years for the inspection and evaluation of the steel components of the intake structure. Describe the basis for the frequency of the structural performance assessments at CCNPP and describe the attributes of the aging management program as it relates to RG 1.127 for steel components.

Section 3.3A.2 - Primary Containment Structures

11. Describe the past performance experience of the permanent pipe drain system for the primary containment structure foundation. Please provide the basis for concluding that most of the predicted settlement is in terms of uniform settlement, for any previously experienced cracking of the concrete basemat, degradation, deformation or excessive straining of the containment dome, wall and basemat.

12. Provide a summary description of the Time-Limited Aging Analysis (TLAA) that will be performed for the three types of containment prestressing tendons and explain the basic assumptions and limitations that will be used in the evaluation.

13. Are the transfer tube/bellows and containment sump recirculation penetrations accessible for periodic inspections? If not, discuss the rationale for concluding that the

functionality and integrity of these items are assured and maintained during the license renewal period.

14. Provide a discussion of how the following degradation mechanisms were determined to be non-plausible for the CCNPP primary containment structure: (a) scaling, cracking and spalling of concrete dome, wall and basemat, and loss of bond and material of embedded steel and reinforcement; (b) cracking, distortion, component stress level increase, and loss of strength and modulus due to elevated temperature of the concrete basemat; and (c) corrosion and loss of prestress of hoop and dome tendons.

15. Since 1997, CCNPP Units 1 and 2 have experienced degradation in their containment prestressing systems. Provide a description of the aging effects associated with this degradation and aging management program(s) that will be relied on to manage these aging effects for the period of extended operation.

16. Provide a discussion of how STP-M-663-1 and STP-M-663-2, in conjunction with the proposed lift-off force TLAA, will ensure that the effects of tendon corrosion and loss of prestress force are adequately managed. Describe how items such as: (a) preventive actions will mitigate or prevent aging degradation; (b) aging effects will be detected before losing structure and component function; (c) measures incorporated in the procedure will effectively reflect past CCNPP operating experience with respect to tendon corrosion and loss of prestress and eliminate the root causes identified during post tendon degradation assessment; and (d) timely detection of aging effects and corrective action implementation are fully realized.

17. Provide a discussion of how STP-M-663-1/2 surveillance procedures effectively manages the potential additional tendon force loss (8 to 14 %) due to elevated temperature resulting from abnormal sun exposure or proximity to hot penetrations (refer to NUREG-1611, page 18, issue 14).

18. Referring to page 3.3A-15, fourth paragraph, it is stated that ,"...Other intended function (structural or functional support to safety-related (SR) equipment, shelter/protection of SR equipment, and missile barrier) will not be affected because those functions will be provided by the containment wall itself." Clarify this conclusion with consideration that the design strength of the containment wall is dependent on the availability of the prestress level prescribed in the design analysis calculations and any reduction or deviation of the actual prestress level in a wall section from that of the designed prestress level will reduce both the strength and the margin of the wall, which may lead to loss of wall integrity and functionality.

19. Are there any parts of the primary containment structures 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: (1) Preventive actions that will mitigate or prevent aging degradation; (2) Parameters monitored or inspected relative to degradation of specific structure and component intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

20. Referring to the plausibility of microbiologically-induced corrosion of fuel transfer tube, provide an explanation for concluding that the stress level of the CCNPP fuel transfer tube is lower than the threshold to cause the microbiologically-induced corrosion.

21. Provide the justification for the inspection frequencies in CCNPP procedures MN-3-100, MN-1-319, STP-M-665-1/2 and STP-M-661-1/2, and discuss how they compare to related industry standards including that of the "Rules for Inservice Inspection, Section XI, ASME Boiler and Pressure Vessel Code," and justify any deviations.

22. Explain and justify the modification of CCNPP Administrative Procedure MN-1-319 pertaining to the "authority to deviate from scope or schedule" described on page 3.3A-24 of the application.

23. Provide a summary discussion of the method and procedures used in the 1992 containment inspection including a list of deficiencies found. Describe how the experience from the inspection was incorporated into the proposed revision of the walkdown procedure MN-1-319, as applicable. In addition, clarify the basis upon which you concluded that the components in the containment system were in good to excellent condition.

24. Provide a justification for excluding from the aging management review that part of the liner that is embedded horizontally inside the concrete basemat from the aging management review, and discuss how the aging effects of this part of the liner will be managed to ensure its functionality for the extended period of operation. It appears that the embedded horizontal basemat liner, because of its relatively low elevation and horizontal orientation, tend to have a higher likelihood of water accumulation/retention on its surfaces, which in turn, might result in a higher potential for liner corrosion/degradation. Discuss how this specific concern as well as any other applicable aging effects are factored into your liner aging management program.

25. Provide a justification for determining corrosion and degradation of the concrete shell-side liner surfaces and the anchor studs is not plausible. It is recognized that due to the presence of prestressing forces on the shell concrete, there will be a lesser degree of moisture penetration through the concrete to reach the liner surfaces and the anchor studs. However, it is not totally clear to the NRC staff that the concrete shell-side liner surface and anchor stud corrosion can be determined to be non-plausible. If available,

please provide a description of the results actually observed from concrete side liner surface examination to support your non-plausibility conclusion.

26. How did BGE consider Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," in the context of license renewal? Describe your plans for participating in any industry efforts in preparing the response to the generic letter as it relates to license renewal.

Section 3.3B.2, Turbine Building Structure

27. Provide the basis for excluding the seismic Category II portions of the turbine building from consideration in addressing Intended Function No. 5. Was consideration given to the potential for adverse impacts on the SR structures, systems and components within the turbine building if aging related degradation results in the turbine building, which is not a seismic Category I structure, being damaged or collapsing under a design basis event? Also, discuss how the venting functions will be maintained if the siding and retainer clips are not classified as SR. (reference Section 3.3B.1, pp 3.3B5-6)

28. Regarding Structure Description/Conceptual Boundaries (p.3.3B-2), BGE states that "The circulating water intake and discharge conduits are incorporated into the spread footings."

    a. Do these conduits perform any of the seven identified intended functions?

    b. Are the conduits classified as SR? If not, describe their design standards.

    c. Are conduits subject to any aging management review? If so, where in the license renewal application (LRA) are these conduits addressed? If not, justify why they were excluded.

    d. Provide a summary description, including the important elements, of BGE's current and future program for managing aging effects on these conduits.

29. Address the following questions related to Table 3.3B-2 and Table 3.3B-3 regarding seismic Category I electrical duct banks:

    a. Provide a summary description of how the seismic Category I conduits were encased in the ductbanks. Are a number of conduits individually encased in concrete or are a number of conduits collectively routed through void spaces under the turbine building?

    b. What is the chain of events that may lead to water seepage into the conduits?

c. What are the consequences of water seepage into the conduits? How would this affect the power cables to the saltwater pumps?

d. Why is intended function No. 2 not affected by water seepage into the conduits? Explain this apparent inconsistency with Table 3.3B-2.

e. What is the basis for concluding that there are no plausible age related degradation mechanisms (ARDMs) for the ductbanks, relating to the possibility for flowing groundwater?

f. Provide a summary description, including the important elements (such as schedule for inspection, methods, criteria, etc.), of BGE's current and future aging management program for the ductbanks.

g. Address the effects of settlement on the ductbanks.

30. In the last paragraph of Section 3.3B.1, replacement of components is discussed. Provide a description of how this process will be applied, and provide examples of structural components and subcomponents which may be subject to replacement.

31. Are there any parts of the turbine building structures 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: (1) Preventive actions that will mitigate or prevent aging degradation; (2) Parameters monitored or inspected relative to degradation of specific structure and component intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

Section 3.3C, Intake Structure

32. Explain how the actions taken to manage recurring degradations of the structural components (Groups 1, 2, 3, and 4) identified in Section 3.3C.2 during the baseline inspection and subsequent inspections will be integrated into the aging management programs developed for the license renewal term.

33. Figure 3.3C.1 shows the evaluation boundary for the intake structure excludes the intake channel and baffle structure. Are the intake channel and baffle structure within the scope

of license renewal? If not, provide a justification for not including them? If so, where are they addressed in the LRA?

34. Referring to Table 3.3C-2, identify any masonry walls within the scope of license renewal (SR or non-SR whose failure could directly prevent satisfactory accomplishment of any of the required SR functions) in the intake structure? If any intake structure masonry walls within the scope of license renewal are identified, identify where they discussed in the LRA. Describe any masonry walls that were excluded from the scope of license renewal and the basis for their exclusion.

35. Table 3.3C-3 identifies the sluice gate as a long lived/passive structure within the scope of license renewal, but does not identify mechanical wear as a plausible ARDM. Provide a justification for excluding mechanical wear as a plausible ARDM.

36. Section 3.3C.2 states that the expansion joints that run along the intake structure floor have experienced age-related degradation in the past. The degradation allowed water seepage up through the joints that required repairs to affected joints. This is an indication that the intake structure concrete floors, walls, and joints may be exposed to groundwater. What are the potential consequences of this exposure to ground water with respect to aging degradation of the concrete floors and walls and was that identified as a plausible ARDM for inclusion in the aging management review. If not, provide a justification for this conclusion.

37. The salinity and sulfate content of the Chesapeake Bay surface water as found in 1968-69 is high enough to chemically attack the steel components and sluice gates. Describe the basis upon which you concluded that the concentrations of these attributes have not increased in the last 30 years, and describe how the proposed aging management program would address significant increases if they were to occur in the future.

38. Are there any parts of the intake structure 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: (1) Preventive actions that will mitigate or prevent aging degradation; (2) Parameters monitored or inspected relative to degradation of specific structure and component intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

Section 3.3D, Miscellaneous Tank and Valve Enclosures

39. A 1994 inspection of the No. 12 CST and No. 21 fuel oil storage tank enclosures identified minor surface corrosion on steel beams. This surface corrosion was deemed insufficient to affect the structural integrity of the enclosures. Provide a justification for this conclusion, and discuss how the aging management review assessed the structural integrity of the enclosures.

40. Has the auxiliary feedwater valve enclosure been previously inspected for corrosion of steel components or degradation of protective coatings? If so, provide a summary of the results.

41. Provide a description of the amount of corrosion or degradation of protective coatings that will be allowed on tanks and valve enclosures before corrective action is implemented. If degradation is observed, what will be the acceptance criteria to determine that intended functions will be maintained with a sufficient margin?

42. Are there any parts of the tank and valve enclosures 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: (1) Preventive actions that will mitigate or prevent aging degradation; (2) Parameters monitored or inspected relative to degradation of specific structure and component intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

Section 3.3E, Auxiliary Building and Safety-related Diesel Generator Building Structures

43. Section 3.3E appears to address the license renewal aspects of the safety-related emergency diesel generator (EDG) structures, but not the station blackout diesel generator (DG) structure. Figure 3.3E-1, identifying site structures within the scope of license renewal (WSLR), also does not include the blackout DG structure (attached to the EDG 1A building) as being WSLR. Since the blackout DG systems are WSLR according to Section 5.8 of the technical report, identify where in your application the license renewal aspects of the blackout DG structures are discussed. If you have concluded that the blackout DG structures are not WSLR, provide your rationale for that conclusion.

44. Figure 3.3E-1 shows a number of WSLR structures such as the condensate storage tank enclosure, auxiliary feedwater valve enclosure, and the fuel oil storage tank enclosure, that are somewhat physically removed from the systems they support. There are no interconnecting structural tunnels/raceways for piping and cabling shown on this figure. Please identify and describe any interconnecting structures associated with these components and address the corresponding license renewal aspects of these structures, as necessary. Also, address any other interconnecting structures between major buildings/components that are not shown on this figure and describe where the license renewal aspects are addressed.

45. One of the structural functions identified for structures that are WSLR is to provide flood protection barriers for an internal flooding event. Generally, portions of the equipment and floor drainage system (EFDS) may also be relied upon for adequate protection against internal (and sometimes external) flooding. Identify if any of the EFDS associated with the auxiliary building and EDG structures that are relied upon for protection against internal or external flooding. Also, identify where the license renewal aspects of the WSLR portions of the EFDS are addressed. Otherwise, provide justification for your determination that no portions of the EFDS are WSLR.

46. With regard to the discussion on page 3.3E-3, please discusse: (a) the basis for not including the 1-story missile protection structure located on the east side of the Safety-Related Diesel Generator Building (SR-DGB) within the review scope of the SR-DGB, and (b) describe actions taken to support your conclusion that there has been no evidence of age-related degradation for the SR-DGB.

47. Regarding the entry on Table 3.3E-3, first column, "Concrete (Including Reinforcing Steel)," and "Structural Steel," please provide a justification for determining the following mechanisms as not being plausible ARDMs: corrosion of embedded steel/rebar, cracking of concrete/masonry walls, settlement and corrosion of structural steel.

48. With regard to the discussion on page 3.3E-12, please discuss if any maintenance or watertable elevation monitoring programs are in place to ensure proper functioning of the system and what their role would be in the aging management program.

49. The last paragraph of page 3.3E-12 states that "Most of the predicted settlement is expected in terms of uniform settlement." Please describe the results of monitoring the settlement that led to the assessment that the differential settlement is expected to be small and have a negligible effect. If no monitoring has been performed, provide a justification for this statement.

50. With regard to the discussion on page 3.3E-13, please provide a summary description (including scope and findings) of any past or existing inspection program(s) which led you to state that "no cracking or other evidence of settlement that would affect structural integrity has been observed to date."

51. Page 3.3E-18 indicates that one of the objectives of Calvert Cliffs Administrative Procedure MN-1-319, "Structural and System Walkdowns" program is to assess the

condition of the structures, systems, and components such that any abnormal or degraded condition will be identified, documented, and corrective actions taken before the condition proceeds to failure of the structures, systems and components (SSCs) to perform their functions. Please discuss the frequency with which walkdowns of the SSCs will be carried out and the basis for those frequencies.

52. With regard to the discussion on page 3.3E-20, what has been the average leak rate of water from the spent fuel pool (SFP) liners based on past years of observation? If the SFP liner cannot be confirmed as the source of water collected during the monthly testing, indicate what other potential sources of water the leakage observed to-date can be attributed to in the results of the monthly test? Discuss if there are written procedures available which are used to guide the liner walkdown task and ensure its reasonable performance of functions. Also, based on your past experience, have you ever identified any significant corrosion, thinning, or cracking of liner plates? If yes, discuss the corrective actions taken.

53. With regard to the discussion on Page 3.3E-21, is it your conclusion that the conditions necessary for stress corrosion cracking of the SFP liner do not exist supported by actual field observation of liner conditions?

54. Provide a summary description (including operating experience) of the SFP liner performance program including the scope and inspection frequency.

55. Page 3.3E-26 states that "Experiments have shown that the Carborundum sheets can experience spalling and surface abrasion, which result in a loss of boron carbide, ..." Please discuss the extent of actual spalling you have experienced to date. Also discuss the potential for the debris from Carborundum spalling to accumulate in a asymmetrical fashion to the extent that partial clogging of some gaps between the spent fuel rack cells can result in the loss of partial fuel cooling function. What programs and activities are in place to manage the potential accumulation of the debris for the period of extended operation?

56. With regard to the discussion on Page 3.3E-26, please provide a discussion of the modified content of the coupon surveillance program which reflects the reevaluation of the sampling intervals to monitor Carborundum and Boraflex condition through the period of extended operation.

57. Are there any parts of auxiliary building and EDG structures 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: (1) Preventive actions that will mitigate or prevent aging degradation; (2) Parameters monitored or

inspected relative to degradation of specific structure and component intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

Section 6.2, Electrical Commodities

58. Discuss whether corrosion allowances were provided in the design of electrical commodities (EC) and how corrosion is addressed as part of the aging management program.

59. Page 6.2-1 of the application states that operating experience relevant to aging was obtained based on CCNPP specific information and past experience. Please provide a summary discussion of any industry wide operating experience that you concluded was applicable to aging mechanisms for electrical commodities.

60. Page 6.2-2 of the report states that "EC are usually not subject to extreme conditions or excessive loads; however, some CCNPP EC are subject to corrosive environments." Provide a summary description on how the environmental stressors (vibration, heat, radiation, and humidity) and operational stressors (internal heating from electrical or mechanical loading, physical stresses from mechanical or electrical surges, vibration, and abrasive wearing of parts) that have resulted in age related failures in electrical commodities were explicitly addressed in the aging management program(s).

61. Clarify your basis for concluding that the preventive maintenance (PM) program can be relied on to detect electrical stressors, as described on page 6.2-9 of the report.

62. Does the PM program include monitoring and trending? If so, please describe the monitoring and trending activities.

63. Are there any parts of the electrical commodities 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: (1) Preventive actions that will mitigate or prevent aging degradation; (2) Parameters monitored or inspected relative to degradation of specific structure and component intended functions; (3) Detection of aging effects before loss of structure and component intended functions; (4) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (5) Acceptance criteria to ensure structure and component intended functions; and (6) Operating experience that

provides objective evidence to demonstrate that the effects of aging will be adequately managed.

September 9, 1998

PUBLIC DOCUMENT ROOM
'98 OCT -1 A8:

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT (CCNPP) UNIT NOS. 1 & 2, LICENSE RENEWAL APPLICATION, SEVERE ACCIDENT MITIGATION ALTERNATIVES (TAC NOs. MA1524 and MA1525)

Dear Mr. Cruse:

By letter dated April 8, 1998, the Baltimore Gas and Electric Company (BGE) submitted its application for renewal of the CCNPP, Units 1 and 2. As part of the application, BGE submitted an environmental report (ER) prepared in accordance with 10 CFR Part 51. The staff is continuing its review of ER. Based on the review of the information regarding severe accident mitigation alternatives (SAMA) submitted under 10 CFR 51.53(c)(3)(ii)(L), the staff has identified areas where additional information would support the staff's SAMA analysis. These are contained in the enclosure.

Please provide a schedule by letter or telephone for submittal of your response within 30 days of receipt of this letter. Additionally, the staff is willing to meet with BGE prior to submittal of the response to provide clarification of the staff's request for additional information.

Sincerely,

Original Signed By

Claudia M. Craig, Senior Project Manager
Generic Issues and Environmental
Projects Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosure: As stated

Docket Nos. 50-317, 50-318

cc: See attached list

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

September 9, 1998

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

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Generic Issues and Environmental
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Enclosure: As stated

Docket Nos. 50-317, 50-318

cc: See attached list

Request for Additional Information
Calvert Cliffs Nuclear Power Plant (CCNPP) License Renewal Application
Severe Accident Mitigation Alternatives (SAMA) Analysis

1. The Calvert Cliffs Probabilistic Risk Assessment (CCPRA) model on which the SAMA analysis is based is said to be far more advanced than the Individual Plant Examination (IPE) submitted to NRC in December 1993 and slightly more advanced than the Individual Plant Examination of External Events (IPEEE) submitted in August 1997.

   (a) Provide a description of the major differences in models/assumptions between the CCPRA model used for SAMA and that submitted to and reviewed by the NRC, and the impact of these changes on the risk profile. Include a discussion regarding development of the CCNPP Level 3 model.

   (b) Confirm whether any of these changes were made in the Level 2 analysis, since the discussion and references in Section F.3.2 seem to indicate that the NUCAP+ model is based directly on the IPE Level 2 model.

   (c) Describe the independent peer reviews performed on the CCPRA model used for SAMA. Explain the significant results and overall conclusions of those peer reviews and describe how the results were incorporated in the CCPRA on which the SAMA analysis is based.

   (d) Discuss how the risk information from the external event analyses is incorporated within the NUCAP+ model for CCNPP.

2. Explain how the potential for reactor coolant pump (RCP) seal loss of coolant accident (LOCA) was modeled in the CCPRA used for the SAMA analysis. Describe and justify the major assumptions associated with the RCP seal LOCA model.

3. The IPE indicated that the anticipated transient without scram (ATWS) contribution was a significant risk contributor. Provide a discussion on the modeling of ATWS in the CCPRA used for the SAMA analysis. Explain and justify major assumptions associated with the ATWS model, e.g., the fraction of time during power operation with unfavorable moderator temperature coefficient.

4. The potential core damage risk during some shutdown plant operating states can also be as significant as the at-power risk. Provide a discussion on how the shutdown risk is considered in your SAMA analysis.

5. The discussion in Section 4.1.17.2 regarding offsite exposure cost states that the annual offsite exposure risk is 68.63 person-rem, however, a value of 54.2 person-rem is reported in Table F.1-4. Please explain this apparent discrepancy.

Enclosure

6. Section F.1.2.6 identifies numerous offsite costs that were evaluated using MACCS and summed to arrive at the economic impact of an accident, but model input and assumptions are not identified. Please provide the following: (a) a description of the major input/assumptions for modeling economic impacts, (b) a discussion of the treatment of the economic impacts of fission product fallout into the Chesapeake Bay, and (c) a listing of the MACCS input file for CCNPP (excluding weather data).

7. BGE did not include several factors in the treatment of onsite economic costs. First, the onsite property damage costs associated with cleanup and decontamination were not included on the basis that such costs are covered by property damage insurance. The NRC's regulatory analysis guidelines, NUREG/BR-0058, Revision 2, consider a societal perspective in the performance of these analyses and call for the inclusion of these onsite impacts. The insurance payments are transfer payments and should not be considered as an impact because the insurance payments do not involve consumptive use of real resources. Second, BGE did not include replacement power costs as an onsite economic cost on the basis that such costs are unlikely to be incurred in a deregulated energy market. The NRC guidelines state that replacement power costs be included as impacts, albeit the guidance does not consider the implications of deregulation. In the evaluation of SAMAs, the staff will rely on cost estimates developed in a manner consistent with current regulatory guidance. Accordingly, please provide an estimate of the averted onsite costs for each affected SAMA and an updated maximum theoretical benefit based on inclusion of the above costs, and update the net value analyses and SAMA screening accordingly.

8. The meteorological data used for the MACCS calculations was based on measurements taken from January 1, 1993 to December 31, 1993. Explain why 1993 data was used, and justify that the data for 1993 is representative, e.g., by comparing 1993 with data collected over a longer period.

9. Describe the source of the population data for the year 2030 provided in Table F.1-3. Confirm that this data is based on the latest growth projection, and that geographic areas where major growth is anticipated are accounted for in the input file.

10. Explain why evacuation times based on the current population and infrastructure are considered to be representative of conditions during the renewal period. Provide an assessment of the impact that longer evacuation times could have on risk results and SAMA findings.

11. Provide a breakdown of the consequence measures calculated for each release category, including person-rem doses, and costs associated with each economic impact identified in Section F.1.3.2.

12. The latest CCNPP risk study provides the most relevant information regarding plant-specific contributors to core damage frequency and risk, and should be used as the primary tool for identifying potential SAMAs. The information provided in Section 4.0 and Appendix F.2 does not indicate extensive use of the CCNPP risk study to identify potential SAMAs. The following additional information should be provided in this regard:

    (a) corrected references for each SAMA, if needed. Several SAMAs which appear to be highly focussed on plant-specific systems or risk contributors (and which seem to derive from the CCNPP IPE submittal) may be erroneously attributed to an Oak Ridge study (Reference 18 in Appendix F.2).

    (b) a characterization of the leading contributors to core damage frequency (from dominant sequences or sequence groups), large release frequency (from each containment failure mode or accident progression bin), and dose consequences (from each release class) based on the latest risk study. This information should be structured to provide a framework for subsequently demonstrating that SAMAs addressing each of the major contributors have been identified and evaluated.

    (c) a listing of the SAMAs identified to address each of the major risk contributors identified in (a), with emphasis on those SAMAs that were identified based on the CCNPP risk study.

13. Based on Tables F.2-1 and F.2-2, it appears that 24 rather than 25 SAMAs were combined into 9 "new" SAMAs, and 97 rather than 96 of the original SAMAs were designated for further analysis. Several SAMAs are multiple-part and effectively add 8 more SAMAs, bringing the total number of SAMAs subjected to further study to 105. The discussion in Section 4.1.17.3 should be modified to be consistent with the information provided in the tables, if needed.

14. BGE estimated the net value for each SAMA, and eliminated SAMAs with a negative net value from further consideration. All remaining SAMAs were ultimately eliminated using this criteria. Although a sensitivity analysis was performed to determine the effect of a lower discount rate on the study findings, the impact of uncertainties and incompleteness in other areas of the analysis were not addressed, i.e., uncertainties in core damage frequency (CDF), offsite consequences, and cost analyses, and the impact of differences in CDF between Unit 1 and Unit 2, as discussed in Section 4.1.17.1. In previous evaluations, the staff "screened-in" any design alternative estimated to be within a factor of 10 of being cost beneficial in order to account for uncertainties and incompleteness in the analysis, and subjected those alternatives to further evaluation based on deterministic and engineering considerations. In this regard, please provide the following: (a) an assessment of the impact that uncertainties and Unit 1/Unit 2 CDF differences could have on the identification of cost-beneficial SAMAs, (b) a listing of SAMAs that could become cost beneficial when these factors are taken into account, and (c) an engineering argument supporting BGE's implementation decision for each SAMA identified in item b.

15. In general, where values for "Maximum Benefit" and/or "Cost of Enhancement" are provided in Table F.2-2, the basis for those values is described in Appendix F.4. However, this information is missing for many SAMAs (e.g., the bases for the Maximum Benefit estimates for SAMAs 2, 4, and 10, and the bases for the Cost of Enhancement estimates for SAMAs 3, 6, and 9). The basis for all numerical values should be provided in order to clarify the screening that was performed based on the numerical values. Also, wherever a cost estimate is taken from another source, the applicability of the estimate to CCNPP should be addressed. For example, the cost to create a reactor cavity flooding system was estimated at over 8 million dollars based on a TVA estimate for Watts Bar. The applicability of such cost estimates to CCNPP should be addressed since the CCNPP reactor cavity is easily flooded relative to the Watts Bar cavity due to differences in containment layout.

16. Provide the results or a schedule for the results of BGE's evaluation of the three SAMAs that were still being reviewed at the time of the license renewal application submittal (SAMA numbers 49, 66b, and 96).

Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant

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Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant

cc:

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September 2, 1998

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

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'98 OCT -1 P3:14

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR THE SAFETY INJECTION SYSTEM (TAC NOS. MA1108, MA1109, AND M99222)

Dear Mr. Cruse:

By letter dated March 3, 1998, Baltimore Gas and Electric Company (BGE) submitted for review the Safety Injection System (5.15) 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 (NRC) staff review the Safety Injection System (5.15) report to determine if this 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 Safety Injection System (5.15) 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 related to scoping is needed to complete its review. Should the staff have additional information needs related to aging management they will be forwarded under a separate correspondence.

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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Exhibit 18

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REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
SAFETY INJECTION SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.15
DOCKET NOS. 50-317 AND 50-318

Section 5.15.1 - Scoping

1. 10 CFR 54.21 (a)(1) states that valve bodies are passive. Page 5.15-12 identifies 29 device types as having only active functions, solenoid valves being one of these 29 components. The drawing on page 5.15.6 and 5.15.7 show solenoid valve bodies being within the evaluation boundary. Provide a justification for not including the pressure boundary function of solenoid valve bodies as being within the scope of the aging management review.

2. Page 6.3-3 (Rev. 18) of the Calvert Cliffs Nuclear Power Plant (CCNPP) Updated Final Safety Analysis Report (UFSAR) indicates that there is a small drain valve controlled remotely from the Control Room, which is intended to drain any leakage from the reactor coolant system into the safety injection system. Was this drain valve subjected to an aging management review (AMR)? If so, provide a cross reference to where this is addressed in the license renewal application (LRA). If not, provide the basis for exclusion.

3. Page 6.3-14 (Rev. 21) of the CCNPP UFSAR indicates that the containment sump suctions are enclosed by particulate screens. Are these screens included within the AMR? If so, provide a cross reference to where they are addressed in the LRA. If not, provide the basis for exclusion.

Enclosure

September 15, 1998

MEMORANDUM TO: Christopher I. Grimes, Director
License Renewal Project Directorate
Division of Reactor Program Management

PUBLIC DOCUMENT ROOM

'98 OCT -1 P3:43

FROM: David L. Solorio, Project Manager/original signed by RAnand for DSolorio/
License Renewal Project Directorate
Division of Reactor Program Management

SUBJECT: FORTHCOMING MEETING WITH BALTIMORE GAS AND ELECTRIC COMPANY (BGE) ON LICENSE RENEWAL FOR CALVERT CLIFFS NUCLEAR POWER PLANT (CCNPP), UNIT NOS. 1 AND 2

DATE & TIME: Monday, September 28, 1998
1:00 p.m. - 2:00 p.m.

LOCATION: U.S. Nuclear Regulatory Commission
Two White Flint North
11545 Rockville Pike
Rockville, Maryland 20852
Room T-2B3

PURPOSE: To discuss the status of the review of BGE's license renewal application for CCNPP, Unit Nos. 1 and 2.

PARTICIPANTS:*

NRC	BGE
J. Roe, NRR	R. Heibel
D. Solorio, NRR	B. Doroshuk
R. Prato, NRR	
et al.	

Docket Nos. 50-317 and 50-318

cc: See next page

CONTACT: David L. Solorio, NRR
301-415-1973

*Meetings between NRC technical staff and applicants or licensees are open for interested members of the public, intervenors, or other parties to attend as observers pursuant to "Commission Policy Statement on Staff Meetings Open to the Public" 59 Federal Register 48340, 9/20/94.

Exhibit 19

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