19647

October 16, 1998

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 SECOND REVISED NOTICE OF FILING
(CONCERNING RAIs)[1]

Petitioner, the National Whistleblower Center (Center), 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). In addition, the exhibits 20-34 attached hereto, are added to the basis for the two contentions previously filed by the Center.

[1] An earlier version of this Notice was faxed to the parties on October 15, 1998. That document has subsequently been corrected and the earlier fax version should be replaced by this document.

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On the afternoon of October 13, 1998, Petitioner received NRC Staff's Answer to Petitioner's Motion to Vacate and Reschedule the Pre-hearing Conference (October 9, 1998) through the mail.[2]

Petitioner reviewed the list of additional RAIs referenced in the NRC Staff filing and promptly tried to locate each of the documents. Not all of the documents were available to the Petitioner from any other source as of October 15, 1998.[3] However, the following is a list of the additional RAIs which the Center was able to locate a copy of and which are hereby added to form the basis of Petitioner's Motion to Vacate and Re-schedule the Pre-hearing Conference and Petitioner's two contentions.

Exhibit #	**Document Name**	**PDR Received**
20	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (February 19, 1998) (w/attachment);	3/25/98
21	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 6, 1998) (w/o attachment);	8/24/98
22	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 11, 1998) (w/attachment);	8/31/98
23	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 21, 1998)	

[2] Although the certificate of service indicates that it was sent by e-mail, the Petitioner presumes that the failure to distribute the document to the Center by e-mail was a clerical oversight.

[3] For example, after two thorough searches, Petitioner was unable to locate the RAIs relating to its environmental review transmitted by cover letter dated September 9, 1998 at the PDR.

	(w/attachment);	No date[4]
24	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 21, 1998) (w/attachment);	9/14/98
25	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 26, 1998) (w/attachment);	9/22/98
26	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 26, 1998) (w/attachment);	9/28/98
27	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 27, 1998) (w/attachment);	9/28/98
28	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 27, 1998) (w/attachment);	10/1/98
29	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 27, 1998) (w/attachment);	10/1/98
30	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 27, 1998) (w/attachment);	No date
31	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (August 28, 1998) (w/attachment);	No date
32	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 2, 1998) (w/attachment);	No date
33	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 3, 1998)	

[4] These documents are yet to be officially file stamped by the PDR.

	(w/attachment);	10/1/98
34	Solorio to Cruse, "Request for Additional Information for the Review of Calvert Cliffs," (September 3, 1998) (w/attachment);	No date

There are now 35 RAIs known to exist, including the RAI referenced in petitioner's Motion to Vacate the Pre-Hearing Conference (Oct. 1, 1998), the additional eighteen (18) RAIs referenced in petitioner's Notice of Filing (Oct. 7, 1998), and the additional fourteen (14) to date which were discovered by the Center. The RAIs cover almost every major aspect of BGE's license renewal application.

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

Respectfully submitted,

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

Attorneys for Petitioner National Whistleblower Center

October 16, 1998

February 19, 1998

Docket File

P-1P37

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR THE DIESEL FUEL OIL SYSTEM (TAC NOS. M95457, M95458, M99180)

Dear Mr. Cruse:

By letter dated May 23, 1997, Baltimore Gas and Electric (BG&E) submitted for review the Diesel Fuel Oil System (5.7) technical report as attached to the "Request for Review and Approval of System and Commodity Reports for License Renewal." BG&E requested that the Nuclear Regulatory Commission staff review the Diesel Fuel Oil System 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 BG&E applies in the future.

As requested, the staff reviewed the Diesel Fuel Oil System (5.7) 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 BG&E's methodology for meeting the requirements of 10 CFR 54.21(a)(2). Based on review of the information submitted, the staff has identified in the enclosure, areas where additional information is needed to complete the review.

Please provide a schedule for the submittal of your responses within 30 days of the receipt of this letter. Additionally, the staff would be willing to meet with BG&E 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

cc: See next page

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Exhibit 20

DOCUMENT NAME:G:\WORKING\SOLORIO\DFO_RAI.LTR * SEE PREVIOUS CONCURRENCE

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

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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 NUCLEAR POWER PLANT UNIT NOS. 1 AND 2
SYSTEM AND COMMODITY REPORTS FOR LICENSE RENEWAL

5.7 - DIESEL FUEL OIL SYSTEM

1. Page 5.7-1 of the report describes the diesel fuel oil (DFO) system. Please provide a discussion of the pipe sizes within the system and whether corrosion allowances were provided in the piping design.

2. Page 5.7-1 of the report indicates that the DFO system is a Seismic Category I system. Figure 5.7-1 of the report indicates that certain portions of piping up to the isolation valves are within the scope of license renewal but the piping downstream of the isolation valves up to the next anchor are not within scope. Under the current licensing basis (CLB) the entire pipe run, which includes the associated pipe and the next anchor downstream from the isolation valves, should have been analyzed by the Baltimore Gas and Electric Company (BG&E) to determine that the piping could withstand design basis event loads, such as a seismic event. If there is a failure in the remainder of the pipe run or the associated piping anchors, the identified portions of the piping may not be able to perform their intended function under CLB design conditions. Did the BG&E piping analysis under the CLB analyze the pipe segments from the downstream anchors to the upstream anchors of the isolation valves in a single analysis? If so, explain how the piping between the isolation valves and the downstream anchor points will be appropriately addressed for renewal.

3. Page 5.7-2 of the report describes the DFO system. However, the report does not identify a non-safety related line from the No. 21 fuel oil storage tank (FOST) identified in the Final Safety Analysis Report (FSAR) Section 8.4.1.2. Specifically, page 8.4-7 of the FSAR contains a statement that indicates that the enclosure for the No. 21 FOST "also acts as a dike for No. 21 FOST with fuel being supplied by way of a non-safety related line." BG&E, in its February 14, 1997, response to the staff's request for additional information (question number 17), stated that the non-safety related building drain line for the No. 21 FOST is not within the scope of license renewal because it is not on the "Q-List." Please explain whether this line is relied upon to remain functional during and following any design basis events to ensure any of the intended functions delineated by the license renewal rule (10 CFR 54.4(a)(2)). Since this line is designated to be used to supply the diesel generators in the event of No. 21 FOST rupture, explain whether the rupture of the FOST is postulated to occur as a result of any design basis event that would also require diesel generator operation (via No. 21 FOST).

   In addition, page 5.7-3 of the report indicates that the non-safety related line from No. 21 FOST to diesel generating room waste oil collection tank No. 11 is not within the scope of license renewal. Discuss whether there is a potential for draining No. 21 FOST if the non-safety related line should rupture, and if there are any isolation valves in the line. If there are isolation valves in the non-safety related line, discuss whether the

Enclosure

valves and associated upstream piping are within the scope of license renewal.

4. Page 5.7-2 of the report discusses operating experience with the DFO system regarding aging effects. The report indicates that the DFO system has "in general, performed well." However, later on in the report, on Page 5.7-19, an outstanding site "Issue Report" on the degradation of caulking and sealants which could affect the FOST was mentioned. Provide additional plant-specific operating experience related to the aging effects applicable to the DFO system. Also, discuss any NRC generic communications and other industry experience related to aging that are applicable to the DFO system. Further, the report indicates that the No. 21 FOST was inspected during the 1997 refueling outage. Please provide information on the results of that inspection.

5. Page 5.7-2 of the report describes the DFO system. Discuss whether the DFO system is partially supported by the diesel generator building and foundation and identify where these structures will be evaluated for license renewal.

6. Page 5.7-6 of the report indicates that BG&E may elect to replace components for which the aging management review identifies further analysis or examination is needed. BG&E also indicates that the replacements would not be subject to an aging management review for license renewal. The license renewal rule excludes aging management review for replacements which are based on a qualified life or specified time period. However, replacing components based on their condition or performance is not a basis for excluding components from an aging management review. The condition or performance monitoring program, including replacements, is considered an aging management program for license renewal. Please identify the structures and components that will be replaced and therefore excluded from an aging management review for license renewal.

7. Page 5.7-6 of the report indicates that an electrical fuse has only active functions and is not subject to an aging management review for license renewal. The component-level intended function of fuses to provide continuity has been determined by the staff to be passive, as described in the letter to the Nuclear Energy Institute, dated September 19, 1997. Explain how BG&E intends to address aging management for fuses.

8. Table 5.7-1 (Page 5.7-7) in the report indicates that the FOST for the new emergency diesel generator (EDG) is not included in this technical report. Please identify in which report the FOST and supporting components for the new EDG will be addressed for license renewal. If BG&E has determined that this FOST and supporting components are not within the scope of license renewal, provide the justification for that determination and describe the extent to which the new EDG is relied upon to satisfy the station blackout rule.

9. Pages 5.7-10 and 5.7-19 of the report describe plant procedures MN-3-100, "Paint and Other Protective Coatings," PEG-7, "System Walkdown," and QL-2-100, "Issue Reporting and Assessment Procedure," for managing aging of the Group 1 and 4 components, respectively, for license renewal. Please expand on the summary description for PEG-7 and provide summary descriptions for MN-3-100 and QL-2-100.

The summary descriptions should provide information addressing the specific elements described in Subsection II.C of Section 3.0 of the working draft standard review plan for license renewal (SRP-LR) dated September, 1997. For example, the summary description should include brief information on the operating experience of these programs regarding aging detected in the Group 1 and 4 components, extent of degradation when detected, frequency of occurrence, and resulting corrective actions. Additional examples of what the summary description should include are: inspection frequency, outline of inspection procedures, techniques used, acceptance criteria, assessment and reporting requirements, and guidelines for corrective actions.

10. Page 5.7-12 of the report indicates that BG&E will develop a new program for buried pipe inspection for license renewal. Please provide a summary description of this program addressing the specific elements described in Subsection II.C of Section 3.0 of the working draft SRP-LR. For example, the summary description should include frequency of inspection, consideration for variations in environmental conditions, guidelines for selecting representative samples, inspection techniques, acceptance criteria, assessment and reporting requirements, and guidelines for potential corrective actions.

11. Page 5.7-12 of the report indicates that the existing cathodic protection program is not necessary for license renewal for buried piping. In addition, page 5.7-18 of the report indicates that the FOST bottoms are not subject to any applicable aging effects. BG&E's basis for this conclusion is that the tank bottoms are coated, set on oil-soaked soil, sealed with grout, and protected by cathodic protection. BG&E provided the same basis in its February 14, 1997, response to the staff's request for additional information (question number 5). However, the staff concludes that the aging effects are applicable for license renewal even if there are preventative or mitigation programs to manage those aging effects and the cathodic protection program constitutes an aging management program.

    Accordingly, please identify the applicable aging effects for the FOST bottoms. Describe the aging management programs for the buried piping and FOST bottoms, including the cathodic protection program, that will ensure effective control of the applicable aging effects during the period of extended operation. In particular, please provide a summary description of these programs addressing the specific elements described in Subsection II.C of Section 3.0 of the working draft SRP-LR.

12. Page 5.7-15 of the report describes plant procedures PEO-0-023-2-O-M, "Drain Water From 11 & 21 FOST," CP-226, "Oil Receipt Inspection and Fuel Oil Storage Tank Surveillance," and CP-973, "Determination of Particulate Contamination in Diesel Fuel Oil," for managing aging of FOST internal surfaces for license renewal. Please expand on the summary descriptions for these programs addressing the specific elements described in Subsection II.C of Section 3.0 of the working draft SRP-LR. For example, the summary description should include brief information on the operating experience of these programs regarding water collected and out-of-specification fuel oil found in FOST, extent of deviation from specification when detected, frequency of occurrence, and resulting corrective actions. Additional examples of what the summary description

should include are: inspection frequency and its basis, acceptance criteria, assessment and reporting requirements, and guidelines for corrective actions. Further, please identify the corrosion inhibitor added to the fuel oil, corrosion effects being controlled by the inhibitor, and provide the basis for the effectiveness of the inhibitor in controlling corrosion.

13. Pages 5.7-15 and 5.7-16 of the report indicate that aging management programs are based on specific national codes and standards and industry guidelines. Please expand on how these referenced documents are relied on for aging management and identify the specific portions of these documents. Include the document titles and dates or editions for those referenced documents.

14. Page 5.7-16 of the report indicates that BG&E will develop a new program for the FOST internal inspection for license renewal. Please expand on the summary description of this program addressing the specific elements described in Subsection II.C of Section 3.0 of the working draft SRP-LR. For example, the summary description should include frequency of inspection, acceptance criteria, and guidelines for corrective actions if degradation is found.

15. Page 5.7-19 of the report indicates that BG&E will develop a new program for caulking and sealant inspection for the FOST for license renewal. Please expand on the summary description of this program addressing the specific elements described in Subsection II.C of Section 3.0 of the working draft SRP-LR and explain the extent to which this program is relied upon for other structures and components. For example, the summary description should include guidance for baseline inspections, inspection techniques, acceptance criteria, and guidelines for potential corrective actions.

16. Table 5.7-3 (Page 5.7-21) of the report lists aging management programs for the DFO system for license renewal. However, the list does not include procedure MN-3-100, "Paint and Other Protective Coatings," which is credited for managing the Group 1 and 4 components as described in the text of the report. Please correct Table 5.7-3 to include MN-3-100 for consistency with the text or explain the differences.

August 6, 1998

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR RADIATION MONITORING SYSTEM (TAC NOS. M95455, M95456, M99179)

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'98 AUG 24 P3:5?

Dear Mr. Cruse:

By letter dated May 23, 1997, Baltimore Gas and Electric (BGE) submitted for review the Radiation Monitoring System (5.14) integrated plant assessment technical report as attached to the "Request for Review and Approval of System and Commodity Reports for License Renewal." BGE requested that the Nuclear Regulatory Commission staff review the Radiation Monitoring System (5.14) 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 staff has reviewed the Radiation Monitoring System (5.14) integrated plant assessment technical report against the requirements of 10 CFR 54.21(a)(1), 10 CFR 54.21(a)(3). By letter dated April 4, 1996, the staff approved BGE's methodology for meeting the requirements of 10 CFR 54.21(a)(2). Based on a review of the information submitted, the staff has identified in the enclosure, areas where additional information is needed to complete its review.

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

Sincerely,

Original Signed By

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

Docket Nos. 50-317, 50-318

Enclosure: As stated

cc w/enclosure: See next page

Exhibit 21

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*See previous concurrence

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Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
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RLatta (RML1)
DMartin (DAM3)
WMcDowell (WDM)
DSolorio (DLS2)
SStewart (JSS1)
PDLR Staff

August 11, 1998

Mr. Charles H. Cruse, Vice President
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1650 Calvert Cliffs Parkway
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'98 AUG 31

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 & 2, INTEGRATED PLANT ASSESSMENT REPORT FOR THE COMPONENT COOLING SYSTEM (TAC NOS. M99583, M99584, M99205)

Dear Mr. Cruse:

By letter dated July 30, 1997, Baltimore Gas and Electric (BGE) submitted for review the Component Cooling System (5.3) 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 staff review the Component Cooling System (5.3) 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 staff has reviewed the Component Cooling System (5.3) integrated plant assessment technical report against the requirements of 10 CFR 54.21(a)(1), 10 CFR 54.21(a)(3). By letter dated April 4, 1996, the staff approved BGE's methodology for meeting the requirements of 10 CFR 54.21(a)(2). Based on a review of the information submitted, the staff has identified in the enclosure, areas where additional information is needed to complete its review.

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

Sincerely,

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

Docket Nos. 50-317, 50-318

Enclosure: As stated
cc w/enclosure: See next page

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Exhibit 22

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P PDR

Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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Enclosure

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS UNITS 1 AND 2 COMPONENT COOLING SYSTEM INTEGRATED PLANT ASSESSMENT, SECTION 5.3
DOCKET NOS. 50-317/50-318

Section 5.3.1 Scoping

1. In Subsection 5.3.1.1 under Interfacing Systems the report lists interfacing systems with the CC system. Additionally, the report states that "the CC system at the interface may not be within the scope of license renewal. . ." For the CC system piping at the interfaces identified to be outside the scope of license renewal identify the components at the interfaces that maintain the pressure boundary function.

2. Section 5.4 of the compressed air system report identifies that certain air-operated components used with particular systems will be included (i.e., pressure retaining functions) within the individual system and not the compressed air system. Based on our review we could not determine how the pressure retaining functions of air-operated components were addressed in the CC system. As there are a number of air-operated components within the CC system, for example valves CV-3840, and CV-3825 shown in the CC system piping and instrument drawing 8307210017-26 (obtained from the Calvert Cliffs Update Final Safety Analysis Report), please explain your process for how air-operated components within the CC system were addressed in the license renewal application (LRA) (and if the process used is the same for other applicable systems indicate that also). Additionally, for other air-operated components that are within systems other than compressed air system please provide a cross reference to where these components are addressed in the LRA or provide justification for their exclusion with special emphasis given to why a failure of the compressed air pressure boundary that these components maintain will not affect any safety related functions of the systems in which they reside or the compressed air system.

Section 5.3.2 Aging Management

3. 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: (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. (6) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

August 21, 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 REPORT 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) 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) 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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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
AUXILIARY FEEDWATER SYSTEM INTEGRATED PLANT ASSESSMENT, SECTION 5.1
DOCKET NOS. 50-317 AND 50-318

Section 5.1.1 - Scoping

Regarding the structures and components identified as being within the scope of license renewal for the Auxiliary Feedwater System (AFW), the staff has the following questions:

1. The licensee's simplified drawing (Figure, 5.1-1) and the description of the portion of the AFW system that is within the scope of license renewal in Section 5.1.1.2, including the list of the 47 device types on page 5.1-6, were compared with Figure 10-13, "Auxiliary Feedwater - Unit 2," (also labeled drawing 84-312, Rev. 2) obtained from the Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report. Several components identified in Figure 10-13 are not listed as "device types." in Section 5.1.1.2. These components include the local temperature indicators on the two AFW turbines, the steam traps on the piping from the main steam supply lines to each AFW turbine, and the steam supply stop and control valves to each AFW turbine. Nor were the exhaust piping from the AFW turbines to the roof exhausts identified in Section 5.1.

   Are these instruments and components within the scope of license renewal? If not, explain why these instruments and components are not within the scope of license renewal. If so, provide a cross reference to where these components are addressed in the license renewal application (LRA) or explain why the scoping process as described in Section 2.0 of LRA did not identify these components/instruments?

2. The simplified drawing Figure 5.1-1 and the system interface discussion in Section 5.1.1.1 do not provide sufficient detail or are unclear regarding the transitions between the AFW system and the interfacing systems.

   For example, in Section 5.1.1.1, under <u>System Interfaces</u>, the interface for the Main Steam system is defined as "The turbine throttle valves through the governor valves to the turbine inlet." However, Figure 5.1-1 shows the interface for the Main Steam System extending back beyond the throttle valve to a normally closed (and undefined) valve in the Main Steam system. This same question applies to other systems interfaces in Figure 5.1-1, such as Auxiliary Steam.

   Please provide a modified version of Figure 5.1-1 or use markups of other existing plant drawings that have sufficient detail such that the transition (focusing on components when applicable) between the AFW system and its interfacing systems can be ascertained by the staff.

3. Page 5.1-6 includes the list of AFW system device types designated as within the scope of license renewal because they have at least one intended function. Several of these

Enclosure

device types pertain to the monitoring of condensate level in the condensate storage tank (CST) No. 12 and are not included on the simplified drawing.

Please provide a more detailed drawing or an additional description of the CST components that support this intended function. Markups of existing drawings would be an alternative which would probably provide sufficient detail. Additionally, include the level indication piping and components and indicate interfaces with other support systems (if any) in the drawings and/or descriptions provided to the staff.

August 21, 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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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 COMPRESSED AIR SYSTEM (TAC NOS. M99589, M99590, AND M99207)

Dear Mr. Cruse:

By letter dated July 30, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Compressed Air System (5.4) integrated plant assessment technical report as attached to the "Request for Review and Approval of System and Commodity Reports for License Renewal." BGE requested that the Nuclear Regulatory Commission staff review the Compressed Air System (5.4) 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 Compressed Air System (5.4) integrated plant assessment technical report against the requirements of 10 CFR 54.21(a)(1), 10 CFR 54.21(a)(3). By letter dated April 4, 1996, the staff approved BGE's methodology for meeting the requirements of 10 CFR 54.21(a)(2). Based on a review of the information submitted, the staff has identified in the enclosure, areas where additional information is needed to complete its review.

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

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

Docket Nos. 50-317 and 50-318
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REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT
UNITS 1 AND 2
COMPRESSED AIR SYSTEM INTEGRATED PLANT ASSESSMENT, SECTION 5.4
DOCKET NOS. 50-317 AND 50-318

Section 5.4.1 - Scoping

1. A simplified diagram of the compressed air system (CAS), Section 5.4, depicting the portions of the system that are within the scope of license renewal, as was included with other sections in the license renewal application (LRA) that the staff has reviewed was not provided. As a result, the staff is having difficulty gleaning from the CAS report exactly which portions of the CAS are or are not designated to be within the scope of license renewal. The information in the Updated Final Safety Analysis Report (UFSAR) also did not help in this regard. Please provide a simplified diagram depicting the major portions of the system, consistent with the level of detail provided in other system diagrams provided in the LRA, and discuss in more detail exactly where there are any boundaries that separate license renewal non-scope and within-scope portions of the system. If a simplified diagram is not available then another option would be to use the existing plant P&IDs or UFSAR figures for the air systems, and supplement the drawings with a summary description of the boundaries of the CAS in sufficient detail such that the staff will be able to determine which components are within and outside of the scope of license renewal.

The next two requests for additional information (RAIs), Nos. 2 and 3, arose partly because a simplified diagram was not provided to aid the staff in its understanding of the license renewal boundaries of the compressed air system. In developing your response to RAI No. 1, please consider the following questions, in part, as additional guidance related to the level of detail to include, in order to facilitate the staff's understanding of your responses to these RAIs.

2. From your description of the CAS and its intended functions we concluded that all parts of the CAS that maintain the pressure boundary (main header, branch piping, tubing to instruments and actuators, etc.) are within the scope of license renewal, as described in the CAS report, or are to be included in other sections of the LRA. We also concluded that the instrument air, plant air, and saltwater air subsystems were within the scope of license renewal since they are all interconnected. In order to verify our conclusions, please identify if there are any pressure retaining components in the compressed air system whose failure would result in loss of system pressure, and are not considered to be within the scope of license renewal. If there are any such components, provide a summary justification as to why they do not fall within the scope of license renewal.

3. The CAS report indicates that all components of the CAS that support the system functions, with the exception of the fire protection function, are safety-related and seismic Category I. Please provide clarification between the safety-related and non-safety-related interfaces within the CAS to assist the staff with determining which interfaces within the CAS are within and outside the scope of license renewal.

Enclosure

4. In the description of intended functions of the CAS the auxiliary feedwater air subsystem and a containment air subsystem are identified. Briefly describe these subsystems and identify if they are included within the scope of the CAS report. If these subsystems are addressed in other sections of the LRA provide a cross reference to where they are addressed to facilitate the staff's review.

Section 5.4.2 - Aging Management

5. Provide the CAS piping size, piping material, and corrosion allowances.

6. Provide a description of the CAS external environment(s) and include a discussion of any potential aging effects applicable to the external surfaces of the components requiring an aging management review.

7. Describe the extent to which Section XI leak tests and inspections apply to the CAS if at all. If so, provide a brief summary of the results and discuss how the results were considered in identifying plausible aging mechanisms.

8. Pages 5.4-11, 5.4-12 and references 27 thru 36 mention several Calvert Cliffs Nuclear Power Plant surveillance test procedures and administrative procedures, such as STP M-571F-1, STP M-571F-2, STP M-583-1, STP M-583-2, EN-4-102, EN-4-104 and MN-1-102, for managing aging of the Group 1 and 2 components for license renewal. Please provide a summary description of the procedures regarding how their implementation addresses the following elements for their related aging management program(s): (a) The scope of structures and components managed by the program; (b) 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 function; (e) Monitoring, trending, inspection, testing frequency, and sample size to ensure timely detection of aging effects and corrective actions; (f) Acceptance criteria to ensure intended functions; and (g) Operating experience that provides objective evidence to demonstrate that the effects of aging will be adequately managed.

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

August 26, 1998

Sep 22, 1998
4:17 pm

Mr. Charles H. Cruse, Vice President
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS NOS. 1 & 2, COMMODITY REPORT FOR THE FUEL HANDLING EQUIPMENT AND OTHER HEAVY LOAD HANDLING CRANES (TAC NOS. MA0293, MA0294, AND M99212)

Dear Mr. Cruse:

By letter dated October 22, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Fuel Handling Equipment and Other Heavy Load Handling Cranes (2.2) commodity 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 Fuel Handling Equipment and Other Heavy Load Handling Cranes (2.2) commodity 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 Fuel Handling Equipment and Other Heavy Load Handling Cranes (2.2) commodity report against the requirements of 10 CFR 54.21(a)(1), 10 CFR 54.21(a)(3). By letter dated April 4, 1996, the staff approved BGE's methodology for meeting the requirements of 10 CFR 54.21(a)(2). Based on a review of the information submitted, the staff has identified in the enclosure, areas where additional information is needed to complete its review.

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

Sincerely, Original Signed By

David L. Solorio, Project Manager
License Renewal Project Directorate
Division of Reactor Program Management
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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
FUEL HANDLING EQUIPMENT AND OTHER HEAVY LOAD HANDLING CRANES
COMMODITY REPORT, SECTION 3.2
DOCKET NOS. 50-317 AND 50-318

Section 3.2.1 - Scoping

1. Provide the basis for excluding the spent fuel shipping cask wash down pit, a structural component in the spent fuel storage system, and the fuel transfer tube from the scope of license renewal.

2. Section 3.2.1, pages 3.2-1 and 3.2-2, briefly discuss the spent fuel stainless steel storage racks. While the spent fuel storage racks are not specifically identified as subcomponents within the spent fuel storage system that are within scope of license renewal, they are identified as components subject to an aging management review. Please clarify the scoping conclusion for the spent fuel storage racks, and provide a cross reference to where the discussion is provided in the license renewal application (LRA).

3. Section 3.2.1, under New Fuel Storage and Elevator, states that the new fuel elevators are part of the fuel handling system discussed in a subsequent paragraph in Section 3.2. Please explain why the system is called the new fuel storage and elevator system, yet the new fuel elevators are described as not being part of the system.

4. Section 3.2.1, page 3.2-5 includes a statement that there are components in the crane system that are not subject to the guidelines in NUREG-0612 because (1) there is adequate separation between the lift points and safe shutdown equipment, and (2) the load does not qualify as a heavy load. Please provide the distance Baltimore Gas and Electric Company (BGE) considers as adequate separation and the basis? Also, explain how adequate separation between lift points and irradiated fuel is considered when scoping the components in the crane system that are subject to the guidelines in NUREG-0612.

5. Is the spent fuel shipping cask wash down pit reinforced concrete subject to aging management review (AMR)? If not, provide the basis for excluding it from an AMR.

6. Is the spent fuel shipping cask wash down pit stainless steel liner subject to AMR? If not, provide the basis for excluding it from an AMR.

<u>Section 3.2.2 - Aging Management</u>

7. Provide the basis for concluding there are no potential or plausible age related degradation mechanisms (ARDMs) warranting aging management for the fuel transfer tube.

8. The potential and plausible ARDMs for the fuel handling equipment (FHE) and heavy load handling crane (HLHC) systems have been listed in Table 3.2-1 of Section 3.2 of the license renewal application. Fatigue, wear and mechanical degradation/distortion has been considered a plausible degradation mechanism for the wire ropes. When bent over a sheave, a wire rope's load-induced stretch can cause it to rub against the groove, causing wear on the sheave or drum. Discuss the results of your evaluation of the wear of the sheaves and drums in contact with the wire ropes. Also indicate whether or not the sheaves and drums in contact with the wire ropes are subject to an AMR.

9. Indicate why fatigue, wear, and mechanical degradation/distortion are not considered plausible ARDMs for the clips, bolts and stops in the spent fuel cask handling crane, polar crane (PC), and intake structure semi-gantry crane subcomponents. Additionally, include in the response a discussion as to why mechanical degradation/distortion of clips, bolts and stops is not plausible in light of the fact that these cranes are subject to accidental loadings during normal operations as described in Section 3.2 on page 3.2-23.

10. Low cycle fatigue is considered plausible for the PC rails and fatigue has been identified as a potential ARDM for this item. It is stated in Section 3.2 that this ARDM, if unmanaged, could result in unstable crack growth under design loads at the flame-cut hole locations. Discuss your plans for mitigating the potential failure at flame cut holes and the potential fatigue damage in the PC trolley rails and in other FHE and HLHC components where flame cut holes might exist.

11. In Section 3.2.2 of the LRA, Table 3.2-1 lists those FHE and HLHC related structural components and subcomponents (the spent fuel shipping cask stainless steel support platform, IC trash racks stainless steel structural members, spent fuel pool platform stainless steel structural members, spent fuel elevator subcomponent stainless steel structural members, and fuel transfer carriage subcomponent stainless steel structural members) that are subject to the AMR and the potential and plausible ARDMs for these systems. This table also indicates that the aging effects are not plausible for most of these structural components and subcomponents. Provide a summary of the basis upon which you concluded that the aging effects such as pitting/crevice corrosion, elevated temperature, irradiation, stress relaxation, fatigue, wear, mechanical degradation/distortion, corrosion due to boric acid, are not plausible for those structural components and subcomponents.

12. As described in the first paragraph of Page 3.2-3 in Section 3.2.1, during the 1996 Unit 1 outage, four fillet welds connecting structural members on the fuel upending machine in the refueling pool failed due to low-cycle fatigue. After the implementation of corrective actions, BGE concluded that fatigue will not be plausible for these fuel handling equipment subcomponents. Provide the basis for concluding that low cycle fatigue is not plausible aging mechanism for other welds in stainless steel members such as fuel

transfer tube supports, new fuel elevator subcomponents and other components listed in Table 3.2-1.

13. Provide the basis for concluding that (1) only the polar crane rails need to be covered under Group 3 Aging Management, but not other crane rails, and (2) Group 4 Aging Management is applied only to wire ropes, but not to other crane components and subcomponents.

14. Discuss to what extent "loose bolts" (loose bolts at the connection of steel members, loose anchor bolts at cracked reinforced concrete members, etc.) were considered as aging effects for some of the fuel handling equipment and heavy load handling crane systems?

15. Provide a summary of the visual inspection procedures applied for the fuel handling equipment and heavy load handling crane systems, including the scope, method, acceptance criteria, frequency, and documentation. Alternatively, describe the process for establishing these attributes.

16. Provide a summary of the coatings inspection program that is intended to supplement the existing preventive maintenance tasks associated with the load handling equipment. Cite any Steel Structures Painting Council guidance that is used in the coatings inspection procedures. Generally describe the repair practices that are used for degraded coating conditions on the load handling equipment, and summarize the past experience with degraded coatings.

August 26, 1998

50-317

Sept 28, 1998 3:13 p.m.

P

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

Dear Mr. Cruse:

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

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

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

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

~~9809140276~~ 980826
PDR ADOCK 05000317
P PDR

Exhibit 26

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

Section 4.2.1 - Scoping

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

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

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

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

Section 4.2.2 - Aging Management

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

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

Enclosure

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

August 27, 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 SEP 24 P3:1

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS 1 & 2, INTEGRATED PLANT ASSESSMENT FOR SCOPING (TAC NOS. MA2210, MA2211 AND MA2212)

Dear Mr. Cruse:

By letter dated April 8, 1997, Baltimore Gas and Electric (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 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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Exhibit 27

Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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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

TCheng (TMC)
PShemanski (PCS)
THiltz (TGH)
SLittle (SLL)

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

Scoping

Table 3-1 in Section 2.0 of Appendix A to the application lists the Calvert Cliffs Nuclear Power Plant (CCNPP) systems and structures. Appendix A to the application contains the aging management review of certain systems and structures which are within the scope of license renewal. Aging management of many of the systems and structures listed in Table 3-1 are not apparently discussed in the application. These latter systems and structures may not be within the scope of license renewal or may have been addressed as part of other systems and structures. To assess whether Baltimore Gas and Electric Company has identified CCNPP systems and structures within the scope of license renewal, the staff is sampling the following CCNPP systems and structures from Table 3-1, which are not apparently discussed in the application:

CCNPP Designation	
2	Electrical 125VDC Distribution
4	Electrical 4kV Transformers & Buses
5	Electrical 480V Transformers & Buses
48	Engineering Safety Feature Actuation
58	Reactor Protective
	Transformer Foundation
	Switchgear Structure
	Well Water Pump House

For the above CCNPP systems and structures, please provide a brief description of their functions and indicate whether any of the functions are intended functions as defined in 10 CFR Part 54.4. For systems and structures with intended functions, that is, within the scope of license renewal, briefly describe where the system and structure components (such as, bus, insulated cables and connections, insulators, and transmission conductors) are evaluated for aging management in the application.

Enclosure

August 27, 1998

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'98 OCT -1 P2:22

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 WATER CHEMISTRY PROGRAM (TAC NOS. MA1016, MA1017, M99223, MA0601, MA0602, M99227, MA0297, MA0304, M99213, M95453, M[illegible]4, AND M99178)

Dear Mr. Cruse:

By letter dated April 8, 1997, Baltimore Gas and Electric (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 the water chemistry program 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, 50-318

Enclosure: Request for Additional Information

cc w/encl: See next page

Exhibit 28

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Mr. Charles H. Cruse
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Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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WMcDowell (WDM)
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THiltz (TGH)
SDroggitis (SCD)
DSolorio (DLS2)
PDLR Staff

KParczewski (KIP)

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

Water Chemistry Program

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

1. Control of the secondary water chemistry plays an important role in ensuring that steam generators and other components exposed to secondary water will not be damaged by corrosion and will preserve their integrity. Please include the following information on your secondary water chemistry control program:

   a) What amine is being used for controlling pH in the secondary water system?

   b) Specify major differences in the secondary water chemistry (feedwater and/or steam generator) for power operation, startup, and shutdown.

   c) Describe and provide technical bases for any significant differences in secondary water chemistry parameters specified in the BGE CP-217 procedure and the values recommended by the Electric Power Research Institute (EPRI) in their guideline reports, referenced in Section 5.12 of Appendix A to the BGE license renewal application.

   d) Specify the upper limits of the major chemistry parameters and the allowable time period to restore chemistry parameters to acceptable limits.

2. Were there any significant secondary water chemistry excursions in the past? If such excursions have occurred, describe any significant impact on the condition of the plant, such as increased potential for corrosion damage of the components in the secondary water system.

The following questions apply to the primary water chemistry as discussed in Section 4.1, "Reactor Coolant System," and Section 5.2, "Chemical and Volume Control System," of Appendix A to the BGE license renewal application:

3. The scope of BGE Procedure CP-204, "Specification and Surveillance Primary Systems," includes the reactor coolant system (RCS) and the chemical and volume control system (CVCS). These systems perform different functions and consequently have different chemistry procedures. Please describe how CP-204 is applied to the RCS and CVCS.

Enclosure

4. The two factors important to minimize corrosion of the primary coolant system components are pH and lithium hydroxide. Describe the pH level and lithium concentrations during a fuel cycle, or describe the procedure for their control.

50-317

P

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

August 27, 1998

PUBLIC DOCUMENT ROOM

'98 OCT -1 P 3:4

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 COMPONENT COOLING SYSTEM (TAC NOS. M99583, M99584 AND M99205)

Dear Mr. Cruse:

By letter dated July 30, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Component Cooling System (5.3) 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 the Component Cooling System (5.3) 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 has reviewed the Component Cooling System (5.3) 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 1, 1998, the NRC 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 5.3 of BGE's license renewal application, the staff has identified in the enclosure additional areas beyond those outlined in the August 1, 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,

Exhibit 29

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, 50-318
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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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PDLR Staff

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JRajan(JRR)
KParczewski(KIP)
SLittle (SLL)

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NOS. 1 & 2
COMPONENT COOLING SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.3
DOCKET NOS. 50-317 AND 50-318

Section 5.3.2 Aging Management

1. The potential age related degradation mechanisms for the Component Cooling System (CCS) have been identified in Table 5.3-3 of Section 5.3 of the license renewal application. The components of the CCS were judged not to be susceptible to low cycle fatigue or corrosion fatigue. Describe the justification and any specific criteria used to make this determination for the piping, check valves, control valves and the pump/driver assemblies of the CCS.

2. Carbon steel piping bends, elbows and nozzles are vulnerable to erosion corrosion which has been identified as an age related degradation mechanism for the CCS piping. General wall thinning is anticipated as a result of erosion corrosion. Describe the specific evaluations which have been performed (or will be performed) to ensure structural integrity of the piping due to the effects of cyclic fatigue at locations where wall thinning may occur during the extended period of operation.

3. The rate of corrosion of components in the CCS can be mitigated by proper control of water chemistry. Please, provide specifications for water chemistry in the CCS. Your answer should include target values for individual parameters and their monitoring frequency.

4. In a 1996 summary report referenced in Section 5.3.2, several incidents which occurred at Calvert Cliffs that resulted in water chemistry parameters exceeding their action levels were mentioned. Baltimore Gas and Electric Company took actions to correct these conditions. Describe these actions, and the experience gained from implementation of these actions with respect to maintaining water chemistry parameters below action levels.

Enclosure

August 27, 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 REPORT FOR THE EMERGENCY DIESEL GENERATOR SYSTEM (TAC NOS MA1036, MA1037, AND M99218)

Dear Mr Cruse:

By letter dated January 21, 1998, Baltimore Gas and Electric Company (BGE) submitted for review the Emergency Diesel Generator System (5 8) 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 Emergency Diesel Generator System (5 8) 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 Emergency Diesel Generator System (5 8) integrated plant assessment technical report against the requirements of 10 CFR 54 21(a)(1), 10 CFR 54 21(a)(3). By letter dated April 4, 1996, the staff approved BGE's methodology for meeting the requirements of 10 CFR 54.21(a)(2). Based on a review of the information submitted, the staff has identified in the enclosure, areas where additional information is needed to complete its review.

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

Sincerely,

Original Signed By

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

Docket Nos 50-317 and 50-318

Enclosure: Request for Additional Information

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Exhibit 30

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

Section 5.8.1 - Scoping

1. Figure 5-8 1 in Section 5.8.1.1 provides the system boundaries and interfaces for the starting air system and appears to indicate that the check valve upstream of the air receiver is not within the scope of license renewal. Please clarify the location of the interface upstream of the air receiver and provide additional clarification as to which components on either side of the interface are within the scope of license renewal. If the starting air system equipment upstream of the receiver check valve is included in a separate section of the license renewal application please provide a cross reference to the applicable section.

2. If the check valve upstream of the air receiver, as discussed in the previous request for additional information, is not within the scope of license renewal, provide the basis for its exclusion and emphasize how the pressure boundary is maintained at the check valve interface with the air piping it is attached to.

Section 5.8.2 - Aging Management

3. Explain generally how the degradation of tank bottoms is managed, particularly the aging management for the bottom of the diesel fuel oil tanks.

4. Several plants with Fairbanks Morse (FM) emergency diesel generators (EDGs) have experienced problems with degradation of welds in the skid-mounted lube oil and jacket water piping of EDGs during normal operation. Subsequent evaluation showed significant lack of penetration and general lack of quality in the welds, which was believed to have occurred during manufacturing. Since some portions of the piping are subject to vibration induced loads, the potential exists for fatigue failure of welds in the piping during the period of license renewal. Section 5.8.1.2.2 discusses that the skid-mounted piping is not subject to aging management review (AMR) in accordance with 10 CFR Part 54. Discuss the basis for excluding the welds in the jacket cooling water piping beyond the skid from an AMR.

5. Describe the diesel exhaust system at the location where it exits the diesel building. At some facilities, structures surrounding the exhaust components have been damaged by the exhaust gases. Debris from these damaged structures has the potential of blocking the diesel exhaust ducts. If the potential for this condition exists at Calvert Cliffs Nuclear Power Plant Units 1 and 2, provide a discussion of which aging management program is relied on for managing this condition during the proposed extended period of operation.

6. Provide information regarding parameters which will be inspected, monitored and trended for detection of aging effects due to corrosion and fatigue on the internal and external surfaces of the EDG exhaust piping and mufflers. Also provide the acceptance criteria for these parameters.

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

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

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

August 28, 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 SALTWATER COOLING SYSTEM (TAC NOS. MA1020, MA1021, AND M99219)

Dear Mr. Cruse:

By letter dated December 17, 1997, Baltimore Gas and Electric Company (BGE) submitted for review the Saltwater Cooling System (5.16) 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 Saltwater Cooling System (5.16) 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 Saltwater Cooling System (5.16) 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). 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 in future correspondence.

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 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 31

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Mr. Charles H. Cruse
Baltimore Gas & Electric Company
cc:

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Unit Nos. 1 and 2

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LDoerflein (LTD)
BBores (RJB)
SDroggitis (SCD)
RArchitzel (REA)
CCraig (CMC1)
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RCorreia (RPC)
RLatta (RML1)
EHackett (EMH1)
AMurphy (AJM1)
TMartin (TOM2)
DMartin (DAM3)
GMeyer (GWM)
WMcDowell (WDM)
SStewart (JSS1)
THiltz (TGH)
SDroggitis (SCD)
DSolorio (DLS2)
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GHubbard (GTH)
BLeFave (WTL)
SLittle (SLL)

REQUEST FOR ADDITIONAL INFORMATION
CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NOS. 1 & 2
SALTWATER SYSTEM
INTEGRATED PLANT ASSESSMENT, SECTION 5.16
DOCKET NOS. 50-317 AND 50-318

Section 5.16.1 - Scoping

1. According to Figure 5.16-1 and Subsection 5.16.1.1, essentially all of the saltwater system is within the scope of license renewal. The saltwater supply to the circulating water system (CWS) pump seals and the CWS discharge conduits are identified as not within the scope of license renewal. Since Figure 5.16-1 does not show any valves, it is not clear where the interface location is between the portions of the system that are within and outside the scope of license renewal, and how the interfacing locations were chosen. Please identify more clearly the interfaces between the portions of the saltwater system that are within and outside the scope of license renewal and if possible, provide a revised drawing that shows the interface locations more clearly. Please provide the basis for the interfaces that define the portions of the saltwater system that are within and outside the scope of license renewal, and if possible, a more general discussion of the process used for identifying interfaces to assist the NRC staff with its review of other sections of the license renewal application (LRA) where scoping interfaces are discussed.

2. Figure 5.16-1 shows an emergency discharge line coming off the pump discharge header (in lieu of the system discharge header). The NRC staff believes, based on the information provided, that the emergency discharge line coming off the pump discharge would be a safety-related alternative to the system's normal discharge path to the CWS discharge conduits. Please describe the function of this line.

3. Figure 5.16-1 does not show the suction piping to the saltwater pumps. Please identify whether the suction piping is included within the scope of license renewal. If so, provide a cross reference to where the suction piping is addressed in the LRA. Also, identify any strainers and/or screens associated with this system and discuss whether these components are within the scope of license renewal. If so, provide a cross reference to where these components are addressed in the LRA. If either of these components are not within the scope of license renewal provide the basis for their exclusion.

Enclosure

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

September 2, 1998

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

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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Exhibit 32

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JMoore (JEM)
MZobler/RWeisman (MLZ/RMW)
SBajwa/ADromerick (SSB1/AXD)
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)
JStewart (JSS1)
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SDroggitis (SCD)
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PDLR Staff

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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.

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, SECTIONS 4.1, 4.2, 5.2, 5.7, 5.15, AND 5.16 (TAC NOS. MA1016, MA1017, M99223, M99587, M99588, M99206, MA0601, MA0602, M99227, M95457, M95458, M99180, MA1108, MA1109, M99222, MA1020, MA1021, AND M99219)

Dear Mr. Cruse:

By letter dated April 8, 1998, Baltimore Gas and Electric Company (BGE) submitted for review its license renewal application. The Nuclear Regulatory Commission (NRC) staff has reviewed Sections 4.1, "Reactor Coolant System," 4.2, "Reactor Pressure Vessels and CEDMs/Electrical Systems," 5.2, "Chemical and Volume Control System," 5.7, "Diesel Fuel Oil System," 5.15, "Safety Injection System," and 5.16, "Saltwater System," 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
Enclosure: Request for Additional Information
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Exhibit 33

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 NUCLEAR POWER PLANT, UNIT NOS. 1 & 2
INTEGRATED PLANT ASSESSMENT, SECTIONS 4.1, 4.2, 5.2, 5.7, 5.15, AND 5.16
DOCKET NOS. 50-317 AND 50-318

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

1. Discuss whether there are reactor coolant system (RCS) and reactor pressure vessel (RPV) components fabricated from Inconel alloys other than Alloy 600, for example, Alloy 690 and Alloy 800. Discuss whether stress corrosion cracking (SCC) of these components is plausible, including basis for BGE's determination.

2. On pages 4.1-42 and 4.2-27 of the application, BGE indicated that the RCS and RPV components most susceptible to SCC have been or will be replaced. Identify the most susceptible Alloy 600 pressure boundary components and discuss the characteristics that render these components most susceptible to SCC. Describe what material has been or will be used in the replacement components, the schedule for replacement, and the basis for the schedule (i.e., how does the schedule ensure that the components will be replaced before the intended function(s) are compromised). Indicate if the replacement components are or will be within the scope of the Alloy 600 Program.

3. Describe the specific inspection activities for the most susceptible RCS and RPV components under the Alloy 600 program. Include a description of and the bases for the included components, inspection schedules, inspection techniques, inspection procedures, inspection personnel qualification, acceptance criteria, and sample expansion criteria.

4. Describe the most recent example of implementation of BGE's corrective action program initiated by, or related to, the Alloy 600 program. Include a description of the initiating event, the corrective action(s) taken, and how the issue was resolved.

5. The application indicates that the Alloy 600 program will be modified. Describe the reason for the program changes, schedule, and proposed content related to this program modification to include all Alloy 600 RCS and RPV components, including RCS nozzle thermal sleeves.

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

Enclosure

Section 5.2, "Chemical and Volume Control System"

7. Based on the description found on page 5.2-23 of the application, the scope of the Boric Acid Corrosion Inspection (BACI) program appears to be limited to components located inside the containment building. BGE also stated on the same page that the "program also requires examination of specific components for discovery of leakage during each refueling outage." State precisely the scope of Chemical and Volume Control System (CVCS) components in the BACI program and describe how the scope encompasses or bounds all the susceptible CVCS components.

8. Describe how the inspection scope and frequency of the BACI program would detect and correct boric acid corrosion of CVCS components before there is a loss of integrity and component intended functions.

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

10. System walkdowns can identify some aging effects. Explain why Procedure PEG-7, "System Walkdowns," is not explicitly included as part of BGE's aging management program to maintain the CVCS components.

11. A foreign material exclusion program limits the introduction of halogens, loose parts, etc., into the reactor coolant system. Explain why such a program is not explicitly included as part of BGE's aging management program to maintain the CVCS components.

12. Flashing erosion of let down system orifices has been identified at other facilities (Surry and Diablo Canyon). Erosion also occurred downstream of the orifices and compromised welds in the pipe. The BGE application does not address this aging mechanism for the CVCS. Are there similar components at Calvert Cliffs units, and, if so, are there plans for inspection? If so, provide a summary of the inspection plan and schedule.

13. Are there any parts of the systems, structures, and components within the CVCS 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.

14. Page 9.1-31 (Rev. 21) of the Calvert Cliffs Updated Final Safety Analysis Report (UFSAR) indicates that boric acid solution is stored in heated and insulated tanks and is piped in heat-traced and insulated lines to preclude precipitation of the boric acid. If the storage tank and pipe insulation material within the CVCS were subject to an aging management review, identify where they are evaluated in the BGE application. If not, justify why these components have been excluded from the renewal scope.

<u>Section 5.7, "Diesel Fuel Oil System"</u>

15. On Page 5.7-12 of the application, cathodic protection of external surfaces of underground piping is mentioned. However, a statement is made that no credit is taken for the cathodic protection program. National Association of Corrosion Engineers (NACE) International has published Recommended Practice (RP) 01-69 (92), "Control of External Corrosion on Underground or Submerged Metallic Piping Systems," that gives guidance on the protection of underground pipelines. RP 01-69 indicates that coatings and cathodic protection are to be used together. In light of the NACE guidance, clarify BGE's basis for not relying on the cathodic protection program.

16. NACE RP 01-69 also describes methods to determine the effectiveness of coatings and cathodic protection programs. Describe the extent to which BGE includes these methods in its programs.

17. There are additional NACE standards, such as RP0193-93, for managing aging of tank bottoms, such as the fuel oil storage tank (FOST) shell and bottom external exposed surfaces. Discuss the extent to which BGE includes these methods in its programs.

<u>Section 5.15, "Safety Injection System"</u>

18. System walkdowns can identify some aging effects. Explain why Procedure PEG-7, "System Walkdowns," is not explicitly included as part of BGE's aging management program to maintain the Safety Injection System (SIS) components.

19. A foreign material exclusion program limits the introduction of halogens, loose parts, etc., into the reactor coolant system. Explain why such a program is not explicitly included as part of BGE's aging management program to maintain the SIS components.

20. The application describes two instances of water hammer in the SIS that resulted in damage to piping supports. Discuss whether these water hammer events contribute to the aging effects of the SIS components. Also, discuss what corrective actions have been taken to preclude recurrence of water hammers.

21. State precisely the scope of SIS components in the BACI program and describe how the scope encompasses or bounds all the susceptible SIS components.

22. Describe how the inspection scope and frequency of the BACI program would detect and correct boric acid corrosion of SIS components before there is a loss of the structure and component intended functions.

23. Page 5.15-36 indicates that BGE will perform an engineering assessment of SCC for the refueling water tank (RWT) penetrations. Describe the scope of the assessment, and provide the completion schedule.

24. Plant walkdown procedures have been described by both PEG-7 and MN-1-319. As discussed at the meeting on June 26, 1998, clarify the status of the two procedures and describe any significant differences.

25. Table 5.15-1 of the application lists SIS piping with designated "Device Codes" of "-CC," "-DC," "-GC," and "-HC" are subject to aging management review. Please explain these device codes, and describe the piping components in terms of the piping size, piping material, and corrosion allowances.

26. Are there any parts of the systems, structures, and components within the SIS 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.

27. In the report, several plant surveillance test procedures and administrative procedures were mentioned, such as STP M-571G-1(2), STP M-571L-1(2), and CP-204 for managing aging of SIS (Groups 2 and 4) for license renewal. Please provide a summary description of the 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.

Section 5.16, "Saltwater System"

28. If an unacceptable degree of internal pitting were discovered while examining a saltwater system group 1 component, describe how BGE would resolve that condition, in accordance with established procedures, through final disposition. Include a discussion of how augmented inspection might be developed.

29. Provide a summary of the buried piping inspection program as applied to the saltwater system (groups 1 and 2). In the discussion include details of:

    a) inspection scope basis;

    b) inspection methods used;

    c) frequency of inspections and the rationale.

30. Page 5.16-16 of the application indicates that the buried saltwater system piping is protected from corrosion, in part, by an impressed current cathodic protection program. However, the application does not indicate that the cathodic protection program will be relied on to manage aging of buried piping for license renewal.

    Clarify whether BGE relies on cathodic protection for buried piping. If cathodic protection is relied upon for aging management, provide a summary of the program, describe the related inspection and verification activities, and describe corrective measures, if any, resulting from operating experience.

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GGeorgiev (GBG)
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SHou (SNH)
MBanic (MJB)
GHornseth (GPH)

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 CONTAINMENT SPRAY SYSTEM (TAC NOS MA1038, MA1039, AND M99221)

Dear Mr Cruse

By letter dated January 21, 1998, Baltimore Gas and Electric Company (BGE) submitted for review the Containment Spray System (5 6) 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 6 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 5 6 against the requirements of 10 CFR 54 21(a)(1) and 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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Mr. Charles H. Cruse
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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THiltz (TGH)
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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 5.6.2 - Aging Management

1. Section 5.6.2 states that some components in the shutdown cooling (SDC) flowpath experienced significant thermal transients during SDC operations. Please identify these components and characterize the extent of the thermal transients they experienced. Identify the parameters and specific criteria that are used to monitor and manage thermal cyclic fatigue for these components.

2. Section 5.6.2 indicates that core spray (CS) system components in the SDC flowpath, namely SDC heat exchangers, the associated piping, temperature instruments and valves, have fatigue usage factors which are bounded by the fatigue usage of the SDC and safety injection (SI) nozzles that connect the SI system piping to the reactor coolant system (RCS). Clarify the technical justification for this conclusion. Also, describe the fatigue criteria used in the design of the CS system components in the SDC flowpath and justify the applicability of that criteria to the period of extended operation.

3. Section 5.6.2 indicates that based on inservice inspections and additional examinations, it was concluded that the integrity of welds in the CS pump discharge piping and the high pressure safety injection (HPSI) piping from the SDC heat exchanger discharge, have not been affected by the service environment and residual stresses that have induced pipe cracking elsewhere in the industry. It is further stated that, since these portions of the CS system may not have any flow due to flushing or performance testing for periods of at least 30 days during normal reactor operation, they were recognized as portions of the CS system which has a high likelihood of containing stagnant oxygenated borated water, an environmental condition which has induced cracks in welds elsewhere in the industry. On the bases of this information, justify the conclusion that similar cracking of welds due to residual stresses and fatigue will not occur in this portion of the CS system during the period of extended operation.

4. Section 5.6.2 indicates that the SDC and SI nozzles that connect the SI system piping to the RCS are among the 11 fatigue-critical locations selected for monitoring under the Calvert Cliffs Nuclear Power Plant fatigue monitoring program (FMP). Describe the specific criteria used for selecting these nozzles for the FMP and indicate the reason the FMP calls for an engineering evaluation of these nozzles.

5. It is stated in the license renewal application that the FMP monitors and tracks low-cycle fatigue usage for the selected components of the Nuclear Steam Supply System and the steam generators. Describe the parameters that are monitored by the FMP that are applicable to the SDC and SI nozzles in the CS system. Also describe how the monitored parameters are compared to the fatigue analysis of record, and the criteria used to initiate corrective action.

Enclosure

CERTIFICATE OF SERVICE

I hereby certify that the foregoing document was served this October 16, 1998 on the following persons:

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Marian Zobler
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Stephen M. Kohn