CHARLES H. CRUSE
Vice President
Nuclear Energy

Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657
410 495-4455

BGE

September 29, 1997

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Response to the July 25, 1997, Request for Additional Information: License Amendment Request; Change to Reactor Coolant System Flow Requirements to Allow Increased Steam Generator Tube Plugging (TAC Nos. M97855 and M97856)

REFERENCES:

(a) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated January 31, 1997, License Amendment Request; Change to Reactor Coolant System Flow Requirements to Allow Increased Steam Generator Tube Plugging

(b) Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated July 25, 1997, Request for Additional Information - Proposed Technical [Specification] Changes to Reactor Coolant System Flow Limit [], Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M97855 and M97856)

(c) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated June 9, 1997, Revision 1 to the License Amendment Request to Convert to the Improved Technical Specifications (TAC Nos. M97363 and M97364)

1/1

A001

By letter dated January 31, 1997 (Reference a), Baltimore Gas and Electric Company (BGE) submitted a license amendment request to the Nuclear Regulatory Commission (NRC) to support operation of Calvert Cliffs Units 1 and 2 with up to 2500 steam generator tubes plugged in each steam generator. By letter dated July 25, 1997 (Reference b), the NRC requested additional information regarding the license amendment request. This letter provides BGE's response to References (b). In addition, this letter transmits a marked-up page from the recently revised Improved Technical Specification (ITS) license amendment request (Reference c), which was not included in Reference (a) for the reasons explained below.

One of the proposed changes in Reference (a) is a revision to the current Technical Specification Table 4.7-1 to change the maximum allowable lift settings for Main Steam Safety Valves RV-3996/4004, RV-3997/4005, RV-3998/4006, and RV-3999/4007 from 1065 psig to 1050 psig. This change is

necessary to support the reanalysis of the Loss of Load and Loss of Feedwater Events, which credit a more restrictive lift setting range for the highest set valves. In Reference (a), it was noted that this change would not affect ITS, as the ITS submittal proposed relocating this information from the Technical Specifications to the Inservice Testing Program. By Reference (c), BGE revised the ITS submittal. The revised submittal proposes retaining the main steam safety valve lift settings in the ITS as indicated in Table 3.7.1-2. To accommodate this change, ITS Table 3.7.1-2 is marked-up indicating the revised lift settings for the four highest set valves. This change is consistent with those proposed for the current Technical Specifications in Reference (a).

Attachment (1) to this letter provides BGE's response to the questions posed in References (b). Attachment (2) is marked-up ITS Table 3.7.1-2. The information contained in Attachments (1) and (2) does not change the No Significant Hazards Determination presented in Reference (a). Should you have further questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

Charles H. Cruse

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, Charles H. Cruse, being duly sworn, state that I am Vice President, Nuclear Energy Division, Baltimore Gas and Electric Company (BGE), and that I am duly authorized to execute and file this License Amendment Request on behalf of BGE. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other BGE employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Charles H. Cruse

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 29th day of September, 1997.

WITNESS my Hand and Notarial Seal: Denise D. Snukes
Notary Public

My Commission Expires: 2/2/98
Date

CHC/GT/dlm

Attachments (1) Response to NRC Request for Additional Information dated July 25, 1997
(2) Marked-up Improved Technical Specification Page

cc: R. S. Fleishman, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
A. W. Dromerick, NRC
H. J. Miller, NRC
Resident Inspector, NRC
R. I. McLean, DNR
J. H. Walter, PSC

ATTACHMENT (1)

BALTIMORE GAS AND ELECTRIC COMPANY'S
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
DATED
JULY 25, 1997

Calvert Cliffs Nuclear Power Plant
Units 1 & 2
September 29, 1997

ATTACHMENT (1)

BALTIMORE GAS AND ELECTRIC COMPANY
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
DATED JULY 25, 1997

NRC Question

Address concerns associated with reactor coolant pump loop-seal clearing and break orientation, and compliance with the requirements of 10 CFR 50.46(b), including concerns regarding metal/water reaction, and long-term cooling. A non-proprietary submittal by Framatome Technologies Incorporated (FTI) provided additional information to describe the concern and associated phenomena.

The small-break loss-of-coolant accident (SBLOCA) scenario involving the principal safety concern involves long term cooling conditions in which the break size and orientation would be such that the primary system would not depressurize, limiting the emergency core cooling system injection flow rate to that only capable of matching the core boiloff rate due to decay heat. Also in this scenario a column of water in the reactor coolant system pump suction loop seal would inhibit the vent path to the break and exert enough additional pressure to the steam space above the core that the level in the core would be depressed below the top of the core. Should this condition exist as an equilibrium condition, core uncovery would be indefinite, since the attendant decay heat rate would be small and virtually constant. Under these circumstances, the cladding oxidation criteria of 10 CFR 50.46 could be violated.

To address this concern in the near term, the licensee should provide information to assure that the Calvert Cliffs plants, in their present configurations (including plant design, technical specifications, procedures, analyses) will operate such that the criteria of 10 CFR 50.46 will not be violated for its SBLOCA analyses in consideration of the scenario(s) of concern. To confirm its near term assessment and address longer term concerns associated with this SBLOCA scenario, and the capability of this model to quantify the scenario(s) of concern for ongoing operation and future configurations, the licensee should describe its action plan to update its licensing basis SBLOCA model with noding and correlations to explicitly simulate the phenomena for the scenarios) and plant configurations of concern as required by 10 CFR 50.46.

Information supplied by ABB/CE in support of its forthcoming updated SBLOCA evaluation model (S2M) confirms the potential for the scenario(s) of concern for some plant designs, and addresses, both near term and long term, the associated concerns for all designs licensed with the model. This model awaits NRC approval; however, vendor assistance may be available to respond to this question for the model used in the Calvert Cliffs SBLOCA analyses.

BGE Response

Asea Brown Boveri - Combustion Engineering (ABB-CE) is the Nuclear Steam Supply System vendor for Calvert Cliffs. ABB-CE also currently manufactures the fuel for Calvert Cliffs and provides our Emergency Core Cooling System performance evaluation. Reference (1) is the recent ABB-CE response to NRC concerning the Small Break Loss-of-Coolant Accident (SB LOCA) scenario discussed above for all ABB-CE plants.

As discussed in Reference (1), Calvert Cliffs is a "Class A" ABB-CE plant. This is a plant for which the Reactor Coolant System (RCS) loop seal elevation is higher than the top of the active core. Also as discussed in Reference(1), since the Calvert Cliffs loop seal elevation is above the top of the core, Calvert Cliffs will not experience hydrostatically-induced core uncovery due to loop seal clearing and/or refilling behavior. The RCS hydrostatic pressure balance during the SB LOCA scenario of concern is a function of the loop seal elevation. This pressure balance controls the potential for core uncovery. A

plant with a loop seal elevation higher than the top of the core is not susceptible to loop seal clearing induced core uncovery during a SB LOCA. Therefore, the core collapsed liquid level during the time of highest differential pressure will be above the top of the core. As a result, the current Emergency Core Cooling System performance evaluation for Calvert Cliffs is not affected by the above loop seal clearing concern, and the criteria of 10 CFR 50.46 will not be violated.

The Calvert Cliffs Emergency Operating Procedures are based on ABB-CE guidance provided in CEN-152, "Combustion Engineering Emergency Procedure Guidelines." This guidance, with respect to operator response to the SB LOCA scenario of concern, is described in more detail in Reference (1). The Calvert Cliffs optimal procedure for LOCA directs RCS cooldown and depressurization, based on a reasonable time frame for operator actions, within 20 minutes of implementing the LOCA procedure. This rapid, controlled RCS depressurization will significantly increase the High Pressure Safety Injection delivery to the RCS and reduce the inventory loss through the break. Although, as discussed above, the loop seal elevation at Calvert Cliffs will prevent the core collapsed liquid level from falling below the top of the core (after the initial blowdown and recovery) without credit for operator action, the forced cooldown implemented by the operators will avert the conditions that could lead to sustained core uncovery.

As discussed in Reference (1), the ABB-CE SB LOCA evaluation model (the model currently used for Calvert Cliffs) conservatively simulates the phenomena for this scenario and plant configurations of concern as required by 10 CFR 50.46. Under conservative analysis conditions, the calculations presented in Section 5 of Enclosure (1) to Reference (1) demonstrate that the ABB-CE SB LOCA evaluation model adequately calculates loop seal clearing and the hydrostatic pressure distributions that may act to uncover the core during the clearing process including the SB LOCA scenarios of concern. Furthermore, the results of these (Section 5) calculations show that the SB LOCA evaluation model adequately evaluates the impact of break orientation by virtue of its elevation dependent modeling capabilities and separated flow methodologies. Therefore, the current Calvert Cliffs SB LOCA evaluation model uses the noding and correlations that are able to quantify the scenario(s) of concern for ongoing operation and future configurations. As a result, an updating of the Calvert Cliffs licensing basis SB LOCA model is not required to address this concern.

REFERENCE (1): ABB-CE letter to NRC, LD-97-010, "Additional Response to NRC Request for Additional Information Regarding CENPD-137, Supplement 2-P," dated April 3, 1997