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February 28, 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
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 February 13, 1997, Request for Additional Information - Proposed Technical Specification Changes to Reactor Coolant System Flow Limit, Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (TAC Nos. M97855 and M97856)

By 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 Reference (b), the NRC requested additional information regarding the license amendment request. This letter provides BGE's response to the questions posed in Reference (b).

NRC Question

1. *On page 1 of Attachment 1 of the transmittal, it is stated, "The primary effects of plugging Steam Generator (SG) tubes are to reduce SG heat transfer area and increase the Reactor Coolant System (RCS) flow resistance." These effects result in reduced SG steam pressure, reduced RCS flow rate, and increase hot leg temperature. Provide a comparison of key parameters (e.g., RCS*

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pressure, T-average, T-cold, T-hot, steam pressure, SG outlet temperature) between the proposed operation and the design basis analysis.

BGE Response

The following table provides a comparison of the key parameters associated with the license amendment requested in Reference (a):

Normal Operating Pressures/Temperatures:

Parameter	Current Operation	Proposed Operation	Design Basis Analysis[1]
RCS Pressure	2250 psia	2250 psia	2250 psia
T-cold	548°F (max)	548°F (max)	550°F
T-average	571.5°F	574.5°F	577°F
T-hot	595°F	601°F	604°F
SG Steam Pressure	840 psia	790 psia	850 psia
SG Steam Temperature	524°F	517°F	525.2°F

[1] With respect to stresses and fatigue usage.

The proposed normal operating pressure for the RCS, 2250 psia, is unchanged from the current normal operating pressure, and the normal operating pressure used in the design basis analysis with respect to stresses and fatigue usage. This also means that the proposed normal operating pressurizer pressure and temperature remain unchanged.

In addition, the proposed normal operating cold leg temperature is unchanged from the current normal operating cold leg temperature. The maximum allowed cold leg temperature, per Technical Specifications, remains at 548°F. Calvert Cliffs' normal operating cold leg temperature will remain slightly less than 548°F, as required by Technical Specifications. The normal operating cold leg temperature used in the design basis analysis with respect to stresses and fatigue usage is 550°F. Use of a slightly higher normal operating cold leg temperature in the stress and fatigue analysis is conservative with respect to maximizing the calculated usage factors for the affected cold leg components.

Due to the increase in RCS flow resistance and decrease in SG heat transfer area caused by SG tube plugging, T-hot (and T-average) will increase, and SG steam pressure (and steam outlet temperature) will decrease, as compared to current operation. The proposed T-hot and T-average, although slightly higher than the current normal operating T-hot and T-average, are still below the T-hot and T-average used in the design basis analysis with respect to stresses and fatigue usage. Using a slightly higher normal operating T-hot and T-average in the design basis analysis is conservative with respect to maximizing the calculated usage factors for the affected components.

The proposed normal operating steam pressure and steam outlet temperature are less than the current normal operating steam pressure and temperature, and are also less than the normal

operating steam pressure and temperature used in the design basis analysis with respect to stresses and fatigue usage. An evaluation of this difference is discussed in response to Question 2 below.

NRC Question

2. *The decrease in steam pressure results in the increase in the primary-to-secondary pressure difference and break flow. Provide evaluation of the effect of decreased steam pressure on the design basis analysis of SGs with respect to the stresses and fatigue usage.*

BGE Response

The proposed normal operating steam pressure is slightly less than the normal operating steam pressure used in the design basis analysis of the SGs with respect to stresses and fatigue usage. Therefore, the proposed normal operating primary-to-secondary pressure difference is slightly greater than that used in the design basis analysis. The SG tubes and tubesheet are the only components directly affected by the primary-to-secondary pressure difference. The design basis analysis of the tubes conservatively uses a primary-to-secondary pressure difference of 2500 psi in the stress and fatigue calculations. This bounding pressure difference (2500 psi) is based on the worst case transient scenario, and it will not be affected by a decrease in the normal operating steam pressure.

The design basis analysis of the SG tubesheet uses primary-to-secondary pressure differences which could be affected by a lower normal operating steam pressure. The primary-to-secondary pressure differences used in the applied transients in the design basis analysis range from 2250 psi to 0 psi. Since the transients involving normal operating steam pressure have primary-to-secondary pressure differences that are between these governing pressure differences, a change in the normal operating steam pressure from 850 psia to 790 psia will not significantly affect the results of the fatigue analysis. The calculated fatigue usage factor for the limiting location demonstrates that there is more than adequate margin to accommodate this change. Therefore, the proposed normal operating steam pressure is acceptable.

The effect of the increase in normal operating primary-to-secondary pressure difference on break flow during a SG tube rupture has been addressed in a separate evaluation for that event. As discussed in Reference (a), the evaluation of the Steam Generator Tube Rupture Event concluded that the margin to the NRC acceptance criteria for the site boundary dose was adequate to accommodate the slight increase in the site boundary dose caused by SG tube plugging.

NRC Question

3. *On page 2 of the transmittal, you stated that the reduced RCS flow rate results in an increase in the temperature rise (delta-T) between the cold leg and the hot leg. Provide evaluation of the effect of increased delta-T on the reactor vessel and internals, piping, the [Control Element Drive Mechanism (CEDM)] housing, the pressurizer (lower head and upper shell), surge line (stratification), pressurizer spray nozzles, and the reactor coolant pumps, with respect to stresses and fatigue analysis.*

BGE Response

The proposed normal operating temperature rise (delta-T) across the core (53°F) is greater than the current normal operating core delta-T (47°F). However, the proposed normal operating core delta-T is less than the normal operating core delta-T used in the design basis analysis of the reactor vessel and RCS piping with respect to stresses and fatigue usage (54°F). Using a slightly larger normal operating core delta-T in the design basis analysis is conservative with respect to maximizing the calculated usage factors for the affected components.

For the reactor vessel internals, a minimum normal operating coolant temperature (and maximum coolant density) is used in order to maximize the applied loads. Therefore, the design basis analysis for the reactor vessel internals is not affected by the proposed increase in the core delta-T.

For the CEDM housing, a maximum normal operating hot leg temperature of 604°F is used in the design basis analysis with respect to stresses and fatigue usage. The proposed normal operating hot leg temperature is 601°F. It is conservative to use a higher normal operating hot leg temperature in the design basis analysis for the CEDM housing since this maximizes the calculated usage factor.

A separate evaluation of the effect of the increase in the normal operating hot leg temperature (as compared to current operation) on Alloy 600 corrosion of reactor vessel head penetrations (including the CEDM housings) was also performed. As discussed in Reference (a), this evaluation concluded that previous safety assessments to address the consequences of primary water stress corrosion cracking are not affected by the proposed hot leg temperature increase.

As discussed in the response to Question 1 above, the proposed normal operating pressurizer temperature (and pressure) is unchanged as compared to current operation, and to the design basis analysis with respect to stresses and fatigue usage. Since the proposed normal operating cold leg temperature is also unchanged, the pressurizer spray nozzle design basis analysis is not affected. Also, since the proposed normal operating hot leg temperature is slightly higher than current operation, the surge line temperature difference (hot leg to pressurizer) is less. Therefore, the surge line temperature stratification is less pronounced than for current operation, and the associated thermal stresses are reduced. Similarly, since the normal operating pressurizer pressure and temperature are unaffected, the design basis analysis with respect to stresses and fatigue usage for the pressurizer (lower head and upper shell) remains bounding.

Finally, since the proposed normal operating cold leg temperature is unchanged, the design basis analysis for the reactor coolant pumps with respect to stresses and fatigue usage is unaffected.

This additional information does not change the 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 28th day of February, 1997.

WITNESS my Hand and Notarial Seal:

Denise D. Snuhis
Notary Public

My Commission Expires:

2/2/98
Date

CHC/NH/bjd

cc: D. A. Brune, Esquire
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