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410 495-4455

BGE

August 19, 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 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 April 22, 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)

(c) Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated June 9, 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 submitted a license amendment request to the Nuclear Regulatory Commission to support operation of Calvert Cliffs Units 1 and 2 with up to 2500 steam generator tubes plugged in each steam generator. By References (b) and (c), the NRC requested additional information regarding the license amendment request. Attachments (1) and (2) provide Baltimore Gas and Electric Company's response to the questions posed in References (b) and (c), respectively.

A001/1

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 19th day of August, 1997.

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

My Commission Expires: 2/2/98
Date

CHC/NH/dlm

Attachments (1) Response to NRC Request for Additional Information dated April 22, 1997
(2) Response to NRC Request for Additional Information dated June 9, 1997

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
APRIL 22, 1997

ATTACHMENT (1)

BALTIMORE GAS AND ELECTRIC COMPANY'S RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION DATED APRIL 22, 1997

NRC Question No. 1

The consequences of the following accidents must be assessed to support the increase in steam generator (SG) tube plugging from 800 to 2500 tubes per SG.

- *a. Steam Generator Tube Rupture (SGTR)*
- *b. Main Steam Line Break (MSLB)*
- *c. Loss-of-Coolant Accident (LOCA)*
- *d. Rod Ejection*
- *e. Seized Rotor Event*

BGE Response

a. Steam Generator Tube Rupture

In Reference (1), Baltimore Gas and Electric Company (BGE) performed a qualitative evaluation of the effects of the increased SG tube plugging limit to reach the conclusion that the acceptance criteria for the SGTR Event would not be exceeded. Baltimore Gas and Electric Company still believes that the conclusions reached in these evaluations are valid; however, to demonstrate the validity of these conclusions, BGE will re-analyze the SGTR Event quantitatively. It is expected that the re-analysis of this event will require approximately three to four months to complete. This re-analysis is not expected to affect the conclusions presented in Reference (1) for the SGTR or any other Design Basis Event. The re-analysis will also include a determination of the consequences of this event in terms of Control Room operator thyroid dose, per 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 19.

b. Main Steam Line Break (MSLB)

As discussed in Attachment (2), Baltimore Gas and Electric Company is currently re-analyzing this event to ensure its consequences do not exceed the 10 CFR Part 100 limits for site boundary dose or GDC-19 acceptance criteria for Control Room operator thyroid dose. It is expected that this analysis will require approximately three to four months to complete.

c. Loss-of-Coolant Accident (LOCA)

The consequences of the LOCA are analyzed under the Maximum Hypothetical Accident analysis in Section 14.24 of Calvert Cliffs Updated Final Safety Analysis Report. In Reference (1), BGE noted that no additional analyses or evaluations have been performed for the Maximum Hypothetical Accident because plugging SG tubes does not affect the amount of activity or release rates for this event. Therefore, the consequences presented in the Updated Final Safety Analysis Report remain unchanged.

d. Control Element Assembly (CEA) Ejection

As noted below in response to NRC Question No. 2, based on a qualitative evaluation of the CEA Ejection Event, the consequences of this event would not be changed from those previously submitted by BGE, and approved by the NRC.

e. Seized Rotor Event

In Reference (1), BGE re-analyzed the Seized Rotor Event. The resultant site boundary doses for this event were determined to be within the 10 CFR Part 100 guidelines. In Reference (3), BGE provided the results of an evaluation of the Seized Rotor Event, indicating that the GDC-19 limits for thyroid and whole body dose are also met.

NRC Question No. 2

Baltimore Gas and Electric Company must address whether the proposed additional plugging introduces the possibility of a new or different accident than previously assessed, or results in a reduction in the margin of safety. Specifically, the condition which is of concern to the staff is that of overfill of the SG which may be experienced in the tube rupture.

Baltimore Gas and Electric Company has performed accident analyses of the MSLB and the seized rotor. In a response to a question from staff, the licensee indicated that they had performed a qualitative assessment of the impact of tube plugging on the rod ejection and SGTR accidents. Based upon this assessment, the licensee indicated that "the appropriate NRC acceptance criteria for each event were met. Therefore, the analyses of record (including dose assessment) for these events were not revised."

It appears inappropriate for BGE to conclude that, because they determine that the doses are within the staff's acceptance criteria, a revised dose analysis need not be submitted for the staff's review and approval. By that assumption, is the licensee inferring that the analyses for which they are submitting revised assessments exceed the staff's dose criteria? If the licensee concluded that the consequences of the SGTR and the rod ejection were greater than that previously assumed then they would have an unreviewed safety question and the licensee would be required to have those analyses reviewed and approved by the staff.

The requirement for additional analyses is necessary because the additional plugging would result in an increase in the quantity of primary coolant released to the faulted SG in the event of an SGTR. This increase would result in additional releases from the faulted SG. In addition, because more tubes are plugged, the heat removal capability of each SG is diminished. Consequently, it will take longer for the faulted SG to be isolated and for the intact SG to remove the decay heat from the core. For the intact SG, since heat removal capability is decreased, the time to remove decay heat is increased from previous evaluations. Consequently, steam releases from both SGs are likely to be increased from previous evaluations.

For the rod ejection accident, BGE must address whether the additional SG tube plugging results in a greater quantity of fuel melting or in a greater quantity of fuel rods which experienced gap releases. If it does, then the consequences would be increased above those previously analyzed. An unreviewed safety question would exist. The licensee would have to assess the offsite and onsite consequences, and the staff would need to perform a confirmatory analysis and issue a safety evaluation indicating that the consequences of a rod ejection accident remain acceptable with the additional SG tube plugging.

BGE Response

In Reference (1), BGE concluded that the acceptance criteria for the CEA Ejection and SGTR Events would still be met based on qualitative evaluations of the effects of the increased SG tube plugging limit.

As discussed in response to Question No. 1 above, the SGTR Event will be re-analyzed quantitatively. It is expected that the re-analysis of this event will require approximately three to four months to complete. This re-analysis is not expected to affect the conclusions presented in Reference (1) for the SGTR or any other Design Basis Event.

Regarding the CEA Ejection Event, in Reference (1) BGE summarized a qualitative evaluation to reach the conclusion that the margin to the limit on the total average enthalpy of the hottest fuel pellet was adequate to accommodate the slight increase in coolant core exit temperature; therefore, fuel failure would not occur as a result of the CEA Ejection Event. The following discussion provides additional detail to support our conclusions that increasing the tube plugging limit to 2500 tubes per SG will not result in a greater quantity of fuel melting or in a greater quantity of fuel rods that experience gap releases during the CEA Ejection Event.

a. *Does the additional tube plugging result in a greater quantity of fuel melting?*

The analyses of record for the Hot Zero Power (HZP) and Hot Full Power (HFP) CEA Ejection Event are documented in References (4) and (5), respectively. These analyses calculate a peak rod average enthalpy of less than 200 cal/gm (HZP) and 185 cal/gm (HFP). The NRC acceptance limit for fuel melt is 280 cal/gm. Therefore, these analyses demonstrate that no fuel melting occurs.

The generic core physics data used in the References (4) and (5) analyses were reviewed and found to remain applicable to the operating conditions under reduced Reactor Coolant System (RCS) flow. Two of these generic physics parameters (ejected CEA worth and peak) are overly conservative. It is judged that the conservatism in these parameters more than offset the slight adverse effect of reduced RCS flow on the peak rod enthalpy for the transient. In addition, since the CEA Ejection transient is quite rapid (reactor trip occurs in less than five seconds), the core transient is not affected by the changes to SG heat transfer characteristics caused by SG tube plugging. Therefore, the reduced RCS flow and SG tube plugging will not increase the peak rod enthalpy for this event. Since the peak rod enthalpy will not increase, the higher tube plugging limit will not result in a greater quantity of fuel melting.

b. *Does the additional tube plugging result in a greater quantity of fuel rods which experience gap releases?*

The analyses of record for the HZP and HFP CEA Ejection analyses determine that the fuel rods will not experience gap releases based on a fuel failure criterion of 200 cal/gm. During review of the HZP analysis of record (Reference 2) the NRC staff stated in Reference (6) that BGE used a fuel failure criterion that was not acceptable to the staff, and when this is the case, the staff assumes 10% as the amount of failed fuel. Reference (6) requested that BGE confirm that the dose consequences are acceptable when using a fuel failure rate of 10%. Baltimore Gas and Electric Company performed a dose assessment for the HZP CEA Ejection Event to demonstrate the results are well within the 10 CFR Part 100 guidelines when using a fuel failure rate of 10% (Reference 7). The NRC has reviewed and approved BGE's assessment (Reference 8), and concluded that "the 10% failed fuel amount represents a conservative failed fuel estimate for CEA Ejection accidents."

As discussed above, the generic core physics data used in the analyses of record remain applicable to the low RCS flow condition. Two of these generic physics parameters (ejected CEA worth and peak) are overly conservative. It is judged that the conservatism in these parameters more than offset the slight adverse effect of reduced RCS flow on the peak rod enthalpy for the transient. In addition, the core transient is not affected by changes to SG heat transfer characteristics. Therefore, the reduced RCS flow and SG tube plugging will not increase the peak rod enthalpy for this event. Since the peak rod enthalpy will not increase, and since 10% represents a conservative failed fuel estimate for CEA Ejection accidents, the higher tube plugging limit will not result in a greater quantity of fuel rods that experience gap releases.

In addition to the fuel failure assumption of 10% used in Reference (7), the following assumptions were also made for the HZP CEA Ejection dose calculation:

a. 25% of the iodines and 100% of the noble gases generated in the failed fuel are released to the containment.

b. All of the iodines and noble gases in the gap area are uniformly dispersed in the containment

c. 50% of the dispersed iodines become airborne.

d. The maximum containment leakage is 0.2% of the containment volume for the first 24 hours.

e. The exclusion area boundary atmospheric dispersion factor is 1.8E-4 s/m^3.

These assumptions resulted in reported doses of 50 rem thyroid and 1.5 rem whole body, which are well within the 10 CFR Part 100 guidelines (i.e., less than 25% of the guidelines) Therefore, the NRC acceptance criteria for this event are met. These assumptions are also applicable to the HFP CEA Ejection Event.

None of the above assumptions are affected by reduction of RCS flow due to increasing SG tube plugging. Therefore, since tube plugging does not result in a greater quantity of fuel rods that experience gap releases, and since all other assumptions of Reference (7) remain valid, the consequences of the CEA Ejection Event will not increase.

References:

(1) 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"

(2) Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated April 22, 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)

(3) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated March 25, 1997, Second 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)

(4) Letter from Mr. J. A. Tiernan (BGE) to NRC Document Control Desk, dated February 6, 1987, "Request for Amendment; Eighth Cycle License Application"

(5) Letter from Mr. A. E. Lundvall, Jr. (BGE) to Mr. R. A. Clark (NRC), dated October 15, 1982, "Amendment to Operating License DPR-69; Fifth Cycle License Application"

(6) Letter from Mr. S. A. McNeil (NRC) to Mr. J. A. Tiernan (BGE), dated March 12, 1987, "Request for Additional Information"

(7) Letter from Mr. J. A. Tiernan (BGE) to NRC Document Control Desk, dated March 27, 1987, "Unit 2 Cycle 8 Reload - Request for Additional Information"

(8) Letter from Mr. S. A. McNeil (NRC) to Mr. J. A. Tiernan (BGE), dated June 30, 1987, "Revised Safety Evaluation Supporting Amendment No. 108 to Facility Operating License No. DPR-69"

ATTACHMENT (2)

BALTIMORE GAS AND ELECTRIC COMPANY
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
DATED
JUNE 9, 1997

NRC Question No. 1

Please provide a basis for the three-second delay of loss-of-offsite power (LOOP) assumed in the main steam line break (MSLB) analyses described in the amendment request. The original submittal identified that this assumption represents a change in the Calvert Cliffs licensing basis methodology for this event, i.e., is not justified by Calvert Cliffs licensing precedent. The response to this question included in its April 16, 1997 submittal indicated that Baltimore Gas and Electric Company (BGE) knows of no other licensee whose approved MSLB analyses include an assumed delay in LOOP, i.e., the assumption is not justified by precedent for any plant MSLB licensing analyses. The response also attempted to draw an analogy between the delay allowed for certain Combustion Engineering System 80 plant locked rotor analyses and using this assumption for Calvert Cliffs MSLB analyses. There is sufficient dissimilarity between these two events that we do not find this justification acceptable at this time. Use consistent with the guidance given in Standard Review Plan 15.1.5 is one way to justify the LOOP delay assumption. That guidance states:

> *"Assumptions as to the loss of offsite power and the time of loss should be made to study their effects on the consequences of the accident. A loss of offsite power may occur simultaneously with the pipe break, or during the accident, or offsite power may not be lost. Analyses should be made to determine the most conservative assumption appropriate to the particular plant design. The analyses should take account of the effect that loss of offsite power has on reactor coolant pump and main feedwater pump trips and on the initiation of auxiliary feedwater flow, and the effects on the sequence of events for these accidents."*

Experience would indicate that, if the sensitivity study described in the guidance were performed, a LOOP occurring simultaneously with the pipe break would likely be identified as the most conservative assumption for departure from nucleate boiling (DNB)-related criteria.

As identified in the original submittal, a change in the LOOP assumption (including the means of identifying the worst case) represents a model change. As may be inferred from Standard Review Plan 15.1.5 guidance, the change could affect all aspects of the model and its calculated scenarios. Therefore, the entire model, and its performance for all applicable criteria, must be reviewed for approval. Another criterion that must be explicitly addressed in calculations with the model is low temperature overpressure which could occur with primary system cooling, refilling, and repressurization. The methodology must also be used to calculate mass and energy release information to be used in the analyses to address containment conditions and dose criteria for MSLB events.

BGE Response

As noted in Reference (1), a similar assumption has been approved for the Seized Rotor Event for Combustion Engineering System 80 plants. To the best of our knowledge, BGE would have been the first licensee to request NRC approval of a time delay between the reactor trip and the resultant LOOP for the MSLB. However, to expedite review and approval of this license amendment, BGE will revise the MSLB analysis presented in Reference (1). The revised analysis will assume a LOOP concurrent with the reactor trip, i.e., without the three-second delay discussed above. The analysis will ensure the consequences of this event do not exceed the 10 CFR Part 100 site boundary dose limits or the GDC-19 acceptance criteria for Control Room operator thyroid dose. It is expected that this analysis will require approximately three to four months to complete.

NRC Question No. 2

Several places in the April 16, 1997 submittal indicate that a positive moderator temperature coefficient (MTC) was used in the analyses, and refer to a March 28, 1996 letter (Reference 4) as justification for its use. What MTC value was approved? If so, is the value used in the analyses the same as the one that was approved? If not, what is the licensing status of the analysis value? Is that the reason for Reference 4? Discuss use of the positive MTC in analyses and how a conservative MTC is used for all analyses (including anticipated transient without scram [ATWS])

BGE Response

In Reference (2) BGE requested that the maximum allowed positive full power MTC in the Technical Specifications be reduced from $+0.3 \times 10^{-4}$ Δρ/°F to $+0.15 \times 10^{-4}$ Δρ/°F. Reference (3) provided BGE's response to questions asked by the NRC during the review of the License Amendment Request to reduce the maximum allowed positive full power MTC.

Although the Technical Specification change has not yet been approved, BGE has incorporated the more restrictive maximum allowed full power MTC ($+0.15 \times 10^{-4}$ Δρ/°F) in the analysis for the Calvert Cliffs Design Basis Events, where appropriate. By using this MTC value in the Design Basis Event analyses of Reference (1), BGE demonstrated acceptable results for these analyses, and will not be required to re-analyze these events when the Technical Specification change to reduce the full power MTC is approved.

The MTC is measured during the initial start-up after refueling and verified to be within acceptable limits. Current administrative controls require the full power MTC to be less than $+0.15 \times 10^{-4}$ Δρ/°F. The full power MTC then decreases during the course of reactor operation. Therefore, use of a maximum positive full power MTC of $+0.15 \times 10^{-4}$ Δρ/°F in the Design Basis Event safety analyses is conservative. Baltimore Gas and Electric Company does not intend to conduct a revised ATWS analysis with a more positive MTC. Anticipated transient without scram is not a Design Basis Event for Calvert Cliffs, and an ATWS analysis has not previously been performed for Calvert Cliffs using the maximum positive MTC allowed by the Technical Specifications. Please refer to Reference (3) for more details of this subject.

References:

(1) 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

(2) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated March 28, 1996, License Amendment Request: Change to Moderator Temperature Coefficient

(3) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated July 31, 1997, Response to Request for Additional Information Regarding the Technical Specification Change to the Moderator Temperature Coefficient (TAC Nos. M95181 and M95182)