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

Febr... y 20, 1997

SUBJECT: NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING, CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 (TAC NOS. M97855 AND M97856)

Dear Mr. Cruse:

The Commission has requested the Office of the Federal Register to publish the enclosed "Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing." This notice relates to your application for amendment dated January 31, 1997, as supplemented February 13, 1997, which would revise the Technical Specifications to reduce the minimum Reactor Coolant System total flow rate from 370,000 gpm to 340,000 gpm. The proposed changes are necessary to support a larger number of plugged steam generator tubes for future operating cycles.

Sincerely,

/S/

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-317
and 50-318

Enclosure: Notice of Consideration

cc w/encls: See next page

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

February 20, 1997

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

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UNITED STATES NUCLEAR REGULATORY COMMISSION

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NOS. 50-317 AND 50-318

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS

CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License Nos. DPR-53 and DPR-69 issued to Baltimore Gas and Electric Company, for operation of the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, located in Calvert County, Maryland.

The proposed amendment revises the Technical Specifications (TSs) to reduce the minimum Reactor Coolant System (RCS) total flow rate from 370,000 gpm to 340,000 gpm; reduce the Reactor Protective Instrumentation trip setpoint for Reactor Coolant Flow - Low from greater than or equal to 95% to greater than or equal to 92% of design reactor coolant flow; adjust the reactor core thermal margin safety limit lines to reflect the reduced RCS flow rate; and reduce the lift setting range for the eight Main Steam Safety Valves (MSSVs) with the highest allowable lift setting from the current range of 935 to 1065 psig to a more restrictive range of 935 to 1050 psig. In addition to the changes to the TSs necessary to support an increased number of plugged SG tubes, reanalysis of the accident analyses affected by this change identified an Unreviewed Safety Question (USQ) associated with these changes. The USQ results from the determination that the Main Steam Line Break (MSLB) and Seized Rotor Event analyses involve an increased percentage of failed fuel cladding. Finally, three reanalyzed events (MSLB, Loss of Coolant Flow, and Boron Dilution) will require Nuclear Regulatory Commission (NRC) approval due to changes to the methodology or assumptions used to analyze these events.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed amendment defines changes to the operating licenses for Calvert Cliffs Nuclear Power Plant, Units 1 and 2, necessary to support increased steam generator tube plugging. The effects of increased steam generator tube plugging include reduced steam generator pressure and RCS flow rate, and increased core outlet (hot leg) temperature. The Technical Specification changes necessary to account for these effects are reducing the minimum RCS total flow rate from 370,000 gpm to 340,000 gpm; reducing the Limiting Safety System Setting for reactor coolant flow trip function from greater than or equal to 95% to greater than or equal to 92% of design reactor coolant flow; revising the Reactor Core Thermal Safety Limit lines to indicate operation at the lower reactor coolant flow rate; and decreasing the maximum allowable lift settings for the eight highest set Main Steam Safety Valves from 1065 psig to 1050 psig. The Design Basis Events (DBEs) affected by these changes were reanalyzed to determine if the effects of increased steam generator tube plugging, and the associated changes to the Technical

Specifications, could result in exceeding the acceptance criteria applicable to each of these events. Although it was determined that the DBE acceptance criteria would not be exceeded as a result of increased steam generator tube plugging, the analyses for the Main Steam Line Break and Seized Rotor Events indicated an increased percentage of fuel cladding failure as a result of the lower RCS total flow rate; therefore, it was determined that this activity involves a USQ.

Technical Specification 2.1.1 will be changed to establish more restrictive limits on core thermal power and reflect a lower minimum RCS flow of 340,000 gpm. Making the core thermal power limits more restrictive does not initiate a change to plant conditions that would affect other plant components. Therefore, the probability of a previously evaluated accident is not significantly increased. Additionally, the Limiting Conditions for Operation and Limiting Safety System Settings based on these limits remain adequately conservative or will be changed in the Core Operating Limits Report, as appropriate. Therefore, the consequences of a previously evaluated accident are not significantly increased.

Technical Specification 2.2 will be changed to reduce the Reactor Coolant Flow - Low reactor trip setpoint from [greater than or equal to] 95% to [greater than or equal to] 92%, thereby providing additional operating margin to this trip setpoint and the associated pre-trip alarm. Reducing this setpoint does not initiate a change to plant conditions that would affect other plant components. Therefore, the probability of a previously evaluated accident is not significantly increased.

As demonstrated by the revised Loss of Coolant Flow analysis, the proposed Reactor Coolant Flow - Low reactor trip setpoint will continue to provide adequate core protection. A trip setpoint of [greater than or equal to] 92% ensures fuel is not damaged, and the site boundary dose remains a small fraction of the 10 CFR Part 100 guidelines. Therefore, the consequences of a previously evaluated accident are not significantly increased.

Technical Specification 3.2.5.c will be changed to reduce the minimum RCS total flow rate from 370,000 gpm to 340,000 gpm. This change reduces the core heat removal rate and slightly increases the core outlet and average coolant temperatures. This change involves a USQ, as the Main Steam Line Break and Seized Rotor Event analyses have indicated an increase in the number of failed fuel pins during these events as a result of reducing the initial RCS flow rate. The probability of malfunction of equipment important to safety (i.e., fuel pin cladding) during these accidents increases. However, this malfunction is not an accident initiator. Rather, it is a consequence of an accident. Therefore, the probability of a previously evaluated accident is not significantly increased. The consequences of the Main Steam Line Break and Seized Rotor Events are not significantly increased, as the results of the analyses of these events are within the current acceptance criteria established by the NRC.

Analyses and evaluations have been performed to demonstrate that the new flow and temperature conditions are acceptable:

> Fuel and core performance remain within acceptable limits. Analysis and evaluation of fuel mechanical design, core physics, parameters, fuel pin performance, fuel assembly thermal/hydraulic performance, and fuel pin corrosion all demonstrate acceptable results.
>
> The effect of the slightly elevated core outlet and average coolant temperature on the structural integrity of the RCS is acceptable. The RCS penetration inspection program and the steam generator tube inspection program will continue to identify and repair or isolate Alloy 600 cracks prior to inservice failure of these components. The stress analysis for the reactor vessel and piping remain bounding.
>
> The performance of control systems (i.e., feedwater, pressurizer level, and pressurizer pressure) will maintain RCS and steam generator parameters within appropriate limits by periodic adjustment, as necessary. Reactor coolant pump operation will be maintained within acceptable limits by periodic adjustment of the operating curves.

Therefore, the probability of a previously evaluated accident is not significantly increased.

Analyses and evaluations of the DBEs have been performed demonstrating that the NRC acceptance criteria for these events are met. The revised analyses and evaluations consider reduced RCS flow, increased reactor coolant temperature, and increased steam generator tube plugging conditions.

> The results of analyses and evaluations of the Postulated Accidents demonstrate that the site boundary dose is within 10 CFR Part 100 guidelines and the core geometry remains coolable. Loss-of-Coolant Accident analysis results meet the acceptance criteria stipulated in 10 CFR 50.46(b).
>
> The results of analyses and evaluations of Anticipated Operational Occurrences demonstrate that fuel parameters do not exceed the specified acceptable fuel design limits and site boundary dose is a small fraction of 10 CFR Part 100 guidelines. Primary and secondary system pressure remain below the pressure upset limits for the RCS and steam generators, respectively.

Therefore, the consequences of a previously evaluated accident are not significantly increased.

Technical Specification 4.7.1.1. will be changed to reduce the maximum allowable lift setting for the eight Main Steam Safety Valves with the highest lift setpoint. This change will place more restrictive limits on the allowable range of lift settings for these eight valves. The allowable range of lift settings for the proposed change is also allowed by current Technical Specification. Therefore, the probability of a previously evaluated accident occurring is not significantly increased.

The revised safety analyses will credit the highest lift setting for these eight valves as being 1050 psig. The more restrictive limit on the maximum lift setting is required in order to make this Technical Specification consistent with the revised safety analyses. Analyses performed assuming the proposed maximum lift setting for these valves demonstrates that secondary system pressure does not exceed 110% of the system design pressure. Therefore, the consequences of a previously evaluated accident are not significantly increased.

Therefore, operation of the facility in accordance with this amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

The proposed amendment revises limiting parameters to assure safe operation commensurate with the effects of steam generator tube plugging, and will not change the modes of operation defined in the facility license. The analysis of transients associated with steam generator malfunctions are part of the design and licensing bases. This change does not add any new equipment, modify any interfaces with any existing equipment, or change the equipments's function, or the method of operating the equipment. The proposed change does not change plant conditions in a manner which could affect other plant components. Reactor core, RCS, and steam generator parameters remain within appropriate design limits during normal operation.

Therefore, the proposed change could not cause any existing equipment to become an accident initiator.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

The margins of safety associated with this change are defined in the fuel and core-related analyses, the Alloy 600 stress corrosion cracking evaluation, the RCS structural evaluation, the operational evaluation, and in each of the transient and accident analyses affected by the increased steam generator tube plugging.

Reanalysis of the fuel and core-related analyses for fuel mechanical design, core physics, fuel performance, thermal hydraulics, and fuel rod corrosion verified that the fuel and core performance will remain within acceptable limits and will be bounded by the current assumptions for fuel performance in the transient and accident analyses. The Alloy 600 RCS penetration inspection program and the steam generator tube inspection program will continue to find and repair Alloy 600 cracks at the slightly elevated core exit temperature prior to any postulated inservice failure of these components. The stress analyses performed for the reactor vessel and piping remain bounding for the slightly elevated core exit temperature. Additionally, the performance of non-safety-related control systems remains adequate to maintain RCS and steam generator parameters within appropriate operating limits. Therefore, the margins of safety associated with the physical and operational effects of this change will not be significantly reduced.

An evaluation of the affected DBEs confirmed that the established acceptance criteria for specified acceptable fuel design limits, primary and secondary system over-pressurization, 10 CFR 50.46(b), Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, and potential radiation dose during accidents have been completed in support of this license amendment request. The evaluation concludes that, when considering the proposed Limiting Safety System Setting for the Reactor Coolant Flow - Low trip, Limiting Conditions for Operation for RCS total flow rate, and reduced lift settings for eight Main Steam Safety Valves per unit, all applicable acceptance limits are met. Furthermore, the USQ resulting from the reduced RCS total flow rate does not represent a reduction in the margin of safety, as the site boundary dose calculated in the affected DBE analyses is within the current established radiation dose limits and the core geometry remains coolable. Therefore, the margins of safety associated with the transient and accident analyses affected by this change will not be significantly reduced.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the FEDERAL REGISTER a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this FEDERAL REGISTER notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By March 28, 1997 , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Calvert County Library, Prince Frederick, Maryland 20678. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to S. Singh Bajwa: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this FEDERAL REGISTER notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to Jay E. Silbert, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC, 20037 attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated January 31, 1997, as supplemented February 13, 1997, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Calvert County Library, Prince Frederick, Maryland 20678.

Dated at Rockville, Maryland, this 20th day of February 1997.

FOR THE NUCLEAR REGULATORY COMMISSION

Alexander W. Dromerick

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation