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ARTHUR E. LUNDVALL, JR.
VICE PRESIDENT
SUPPLY

November 5, 1982

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Mr. Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 2, Docket No. 50-318
Request for Amendment

REFERENCES:
a) Letter A. E. Lundvall, Jr. to R. A. Clark of November 18, 1980
b) Letter R. F. Ash to D. H. Jaffe of September 20, 1982
c) Letter R. A. Clark to A. E. Lundvall, Jr. of February 5, 1981
d) Letter A. E. Lundvall, Jr. to R. A. Clark of October 15, 1982

Gentlemen:

The Baltimore Gas and Electric Company hereby requests an Amendment to its Operating License No. DPR-69 for Calvert Cliffs Unit No. 2, with the submittal of the enclosed proposed changes to the Technical Specifications.

Proposed Change (BG&E FCR 82-1052)

Delete the following pages of the Unit 2 Technical Specifications and add new pages as indicated:

a) delete page B 7-2 and add new page B 7-2 (Attachment 1)
b) delete page 3-14 and add new page 3-14 (Attachment 2)
c) delete page 3-19 and add new page 3-19 (Attachment 3)
d) delete page 3-21 and add new page 3-21 (Attachment 4)
e) delete page 3-23 and add new page 3-23 (Attachment 5)
f) delete page 3-38 and add new page 3-38 (Attachment 6)
g) delete page 3-39 and add new page 3-39 (Attachment 7
h) delete page 3-41 and add new page 3-41 (Attachment 8)
i) delete page 3-42 and add new page 3-42 (Attachment 9)
j) delete page 7-5 and add new page 7-5 (Attachment 10)
k) delete page 7-5a and b and add new page 7-5a (Attachment 11)

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DISCUSSION AND JUSTIFICATION

The Post TMI-Requirements for Operating Reactors (NUREG-0737 TMI action plan), Item II.E.1.2 specifies the requirements for auxiliary feedwater automatic initiation and flow indication. The NRC Action Plan developed as a result of the TMI-2 Accident, (NUREG-0660, Vol. 1), specifies the requirement for reliability analysis and design modifications for PWR auxiliary feedwater systems. In accordance with the above guidance, we have completed design evaluations and are currently implementing modifications to provide a Third Train auxiliary feedwater capability at Calvert Cliffs Unit 2 and, accordingly, with this submittal, request changes to our Technical Specifications.

The auxiliary feedwater system Third Train modifications consist of a new electric driven pump and associated controls powered from the Emergency Diesel Generator safety bus. Reference (a), as amended by Reference (b), describes the safety grade control logic associated with detecting and mitigating a ruptured steam generator. The system has been designed and modified using the ASME Code without the NA stamp requirement as approved per Reference (c).

In this submittal, we provide some operational flexibility in the Technical Specification action statement to allow a steam driven pump to be out of service for periods not to exceed 30 days. We are confident that such flexibility is justified when considering the probability of a start failure on a steam driven pump concurrent with a passive failure (e.g., pipe break, associated with the cross-connected motor driven trains).

We have deleted the applicability of specification 3.0.4 if certain provisions of the action statement are met. Those provisions referenced in the applicability statement ensure the intent of the LCO is met (i.e. at least two pumps are made **OPERABLE** prior to continued operations).

Items 4.7.1.2.a.1 and 4.7.1.2.a.2 define the surveillance requirements for verifying pump performance. In previous amendments of the Technical Specifications, a minimum secondary steam supply pressure was defined as an initial condition for performing the pump Total Dynamic Head Test. This requirement was only an operational consideration to ensure a minimum steam supply pressure to the steam driven pumps.

In reviewing the Updated Final Safety Analysis Report, steam supply pressure is defined by the minimum pressure to maintain constant pump speed. The actual value referenced in the FSAR is below the corresponding saturation pressure for the minimum temperature limit required in MODE 3 and higher MODES, therefore, we have deleted reference to this limit since we always operate well above the minimum design bases limit. Obviously, this requirement is not applicable for testing the motor driven pumps.

Item 4.7.1.2.a.4 of the surveillance requirements specifies that each AFW flow control valve may be verified in its proper position by observing a setpoint dialed into the flow indicator controller for each valve. The normal position of the AFW flow control valve during MODES 1 thru 3 is open. This is a result of the electro/pneumatic control circuit which modulates valve opening by measuring flow. Verifying the flow indicator setpoint is a more meaningful surveillance versus verifying actual valve position.

Associated with the redesign of the auxiliary feedwater system are changes to the new safe shutdown panel which is being installed and located to meet the requirements of Appendix R (Fire Protection Rule) of 10 CFR 50. This panel, which is located on the 45' elevation of the safety-related switchgear room (immediately adjacent to the Control Room), replaces the existing shutdown panel (2C43) located in the steam driven auxiliary feedwater pump room.

Subpart L.6 of the Appendix for Fire Protection provides that shutdown systems installed to ensure post-fire shutdown capability need not be designed to meet seismic Category I criteria, single failure criteria, or other design basis accident criteria, except where required for other reasons (e.g., because of interface with or impact on existing safety system or because of adverse valve actions due to fire damage). Early in the design stages of the Auxiliary Feedwater modifications a study was completed by our engineering staff to determine the instrumentation required outside the Control Room to effect a safe shutdown in the event of an evacuation. Current emergency procedures require the operator to trip the reactor upon evacuation of the Control Room. This function may be performed in several locations outside the Control Room. Wide range nuclear instrumentation was deleted from the new safe shutdown panel design. This instrumentation has no effect on existing safety system control functions and was considered non-essential in the post trip fire induced safe shutdown scenario since other means outside the Control Room (including status of the control rod drive system and reactor coolant system primary parameter display) were available for verifying the condition of the reactor.

Subsequent to the completion of the design for the new shutdown panel, which is to be installed during the Auxiliary Feedwater modifications, our engineering staff commenced discussions with the Commission on the design bases for the Alternate Safe Shutdown modifications. During these discussions our staff committed to install the wide range nuclear instrumentation on the new shutdown panel. It was not possible to include wide range nuclear instrumentation on the shutdown panel for the Auxiliary Feedwater modifications due to the late stage of the decision to include it. Based on this information it is necessary to remove the wide range nuclear instrumentation from pages 3-38 and 3-39 of the Technical Specifications until reinstallation occurs as a result of the Alternate Safe Shutdown design modifications.

Pages 3-41 and 3-42, Post-Accident Monitoring Instrumentation, has also been amended to reflect changes resulting from the Auxiliary Feedwater Third Train modifications. Table 3.3-10 provides a listing of all instrumentation required to place the plant in a safe shutdown condition following an accident. We have deleted the following instrumentation channels; power range nuclear flux, reactor coolant total flow, and main feedwater flow. Justification for deletion of the power range nuclear flux follows the same line of reasoning and is provided above. Reactor coolant system total flow is considered a non-essential channel of instrumentation in the post-trip condition because of the availability of RCS temperature indication, RCS subcooled margin and Reactor Coolant pumps status. In the absence of adequate core flow, RCS temperature and subcooled margin are indirect indicators of core flow conditions and provide adequate display to ensure appropriate actions are initiated by operations personnel to restore from any abnormal conditions existing in the post-trip condition.

Main feedwater flow is considered a non-essential channel of instrumentation because of the availability of wide range Steam Generator level indication. In the post-trip condition main feedwater flow rate provides no significant information required for restoration of the plant. Steam Generator wide range level, on the otherhand, does provide direct indication of the availability of the main feedwater system to restore adequate water inventories. This indication is backed up, from a functional point of view, with the AFW system feedwater flow rate. For the reasons stated above, we are requesting exemption of the above listed channels of instrumentation from the requirements for post accident monitoring instrumentation.

A discussion of those Design Bases events affected by the safety grade auxiliary feedwater actuation system will be transmitted as a supplement to Reference (d). The Safety Analyses impacted by the safety grade auxiliary feedwater actuation system are: the Loss of Feedwater event, the Feed Line Break event, and the Steam Line Break event. These events were analyzed with and without Loss of AC power on turbine trip, and the Feed and Steam Line Break events were analyzed for the inside and outside containment cases. In addition, a spectrum of break sizes were analyzed for the Feed and Steam Line break events. The results show that the Reactor Coolant System upset pressure limit is not exceeded during Loss of Feedwater/Feed Line Break events. Steam generator water mass is preserved above the minimum acceptable amount required for heat removal capability on all postulated Loss of Feedwater events. Offsite radiation dose limits are within 10 CFR 100 guidelines for all postulated Feed and Steam Line Break events. The minimum post trip Departure from Nucleate Boiling Ratio does not exceed the safety limit analyzed in the current reload application.

In conclusion, we have determined that the following Technical Specification changes do constitute an unreviewed safety question since the consequences of several Design Bases Events previously evaluated are slightly worse and because we propose some slightly less conservative bases for the Technical Specifications per the Enclosure, to our October 15, 1982 reload submittal. However, present analyses do demonstrate that acceptable limits on DNBR, fuel centerline temperature to melt, Reactor Coolant System upset pressure, and 10 CFR 100 site boundary dose rate guidelines would not be exceeded during a Design Bases Event. These changes serve to upgrade the existing auxiliary feedwater system to a more reliable safety grade system in accordance with the current guidance provided in NUREG-0737. These changes to our Unit 2 Technical Specifications should satisfy the requirements of Action Plan Item II.E.1.2.

SAFETY COMMITTEE REVIEW

This proposed change to the Technical Specifications has been reviewed by our Plant Operations and Safety and Off-Site Safety Review Committees, and they have concluded that implementation of this change will not result in an undue risk to the health and safety of the public.

FEE DETERMINATION

We have determined, pursuant to 10 CFR Part 170, Paragraph 170.22, that this Amendment request consists of a Class III amendment for Calvert Cliffs Unit No. 2, and accordingly, we are including BG&E Check No. B179741 in the amount of $4,000.00 to cover the fee for this request.

Should you have further questions regarding this matter, please contact us.

BALTIMORE GAS AND ELECTRIC COMPANY

BY: A. E. Lundvall Jr.
Vice President-Supply

STATE OF MARYLAND :
: TO WIT:
CITY OF BALTIMORE :

Arthur E. Lundvall, Jr., being duly sworn, states that he is Vice President of the Baltimore Gas and Electric Company, a Corporation of the State of Maryland; that he executed the foregoing for the purposes therein set forth; that the statements made therein are true and correct to the best of his knowledge, information, and belief; and that he was authorized to execute the same on behalf of said Corporation.

WITNESS My Hand and Notarial Seal: Ruth H. Grese
Notary Public

My Commission Expires: July 1, 1986
Date

cc: J. A. Biddison, Jr., Esquire
G. F. Trowbridge, Esquire
Mr. D. H. Jaffe - NRC
Mr. R. E. Architzel - NRC