

# BALTIMORE GAS AND ELECTRIC COMPANY

P.O. BOX 1475  
BALTIMORE, MARYLAND 21203

ARTHUR E. LUNDVALL, JR.  
VICE PRESIDENT  
SUPPLY

September 2, 1981

U.S. Nuclear Regulatory Commission  
Region I  
631 Park Avenue  
King of Prussia, PA 19406

Docket Nos. 50-317/50-318

ATTENTION: Mr. Eldon J. Brunner,  
Acting Director, DR&PI

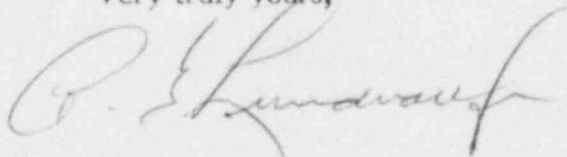
Gentlemen:

This refers to your Inspection Report 50-317/81-04; 50-318/81-04, which transmitted two items of apparent noncompliance with NRC requirements. As a result of additional investigation and discussions held between Mr. D. W. Latham, of Baltimore Gas and Electric Company, and our NRC Resident Inspector, we feel the response to Item I provided by our letter of August 5, 1981, was incomplete. Many of the difficulties associated with this apparent item of noncompliance and our subsequent dialogue have been precipitated by unclear communications and in some cases incomplete information. The information provided by Enclosure (1) to this letter is intended to clarify certain inadequacies which may have existed in our initial reply and to respond to additional questions brought up by our Resident Inspector.

You will note that the information provided by this supplemental reply serves to support our original contention that the Auxiliary Feedwater System remained operable and that the item of noncompliance transmitted by your report was inappropriate. Also, this supplement provides a description of corrective actions we have taken to enhance our communications and review processes, thus ensuring complete and accurate information is used by all parties to preclude misunderstandings such as the one which caused the event in question to occur. We request that you consider this information carefully and reconsider the subject item of noncompliance.

Should you have any questions regarding this reply, we will be pleased to discuss them with you.

Very truly yours,



Vice President - Supply

AEL/DWL/gla

Enclosure (1)

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ENCLOSURE (1)

SUPPLEMENTAL REPLY TO ITEM 1 OF APPENDIX A  
OF NRC INSPECTION REPORT 50-317/81-04; 50-318/81-04

The information provided herein is intended to clarify our previous response submitted on August 5, 1981, and to provide additional information in response to questions which have arisen since that time. This information is broken into two major elements. First, a more detailed description of the results of the calculations conducted to support the Auxiliary Feedwater (AFW) Flow analysis is provided. Secondly, a complete synopsis of correspondence, reports, and documents relating to this subject is supplied in the form of excerpts and comments/corrective actions.

The calculations performed to support the safety analysis of the effects of AFW flow throttling provided the following information:

1. Assuming maximum decay heat in the first hour after shutdown a flow of 440 gpm is required to remove decay heat.
2. After the first hour, a flow of 460 gpm is required to cooldown the Reactor Coolant System (RCS) prior to depleting the contents of the Condensate Storage Tank (CST) dedicated to one unit.

The below items are excerpted from various documents, reports, and correspondence related to AFW flow requirements and the analysis which supported throttling. Where appropriate, comments and corrective actions accompany the excerpted material.

1. The Limiting Condition For Operation Technical Specification 3.7.1.2, "Auxiliary Feedwater System", states:

"3.7.1.2 At least two steam turbine driven generator auxiliary feedwater pumps and associated flow paths shall be **OPERABLE**.

**APPLICABILITY:** MODES 1, 2, and 3.

**ACTION:**

With one auxiliary feedwater pump inoperable, restore at least two auxiliary feedwater pumps to **OPERABLE** status within 72 hours or be in **HOT SHUTDOWN** within the next 12 hours."

2. Technical Specification 1.6 states in part:

"A system, subsystem, train, component or device shall be **OPERABLE** or have **OPERABILITY** when it is capable of performing its specified functions."

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3. Technical Specification Basis 3/4.7.1.2, "Auxiliary Feedwater System", states:

"The operability of the AFW system ensures that the RCS can be cooled down to less than 300<sup>o</sup> F from normal operating conditions in the event of a total loss of offsite power.

Each steam driven AFW pump is capable of delivering a total feedwater flow of 700 gpm at a Total Dynamic Head of 2490 ft to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the RCS temperature to less than 300<sup>o</sup> F when the shutdown cooling system may be placed into operation."

4. Limiting Condition For Operation Technical Specification 3.0.3 states:

"In the event of a Limiting Condition for Operation and/or associated **ACTION** requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the facility shall be placed in at least **HOT STANDBY** within 1 hour and in **COLD SHUTDOWN** within the following 30 hours unless corrective measures are completed that permit operation under the permissible **ACTION** statements for the specified time interval as measured from initial discovery. Exceptions to these requirements shall be stated in the individual specifications."

5. Final Safety Analysis Report (FSAR) 10.2.3 AUXILIARY FEEDWATER SYSTEM, states in part:

"The AFW System, shown in Figure 10-2, 10-2A, and 10-6 (Unit Two Figures 10-9 and 10-9A) is designed to provide feedwater for the removal of sensible and decay heat, and to cool the primary system to 300<sup>o</sup> F in case the main condensate pumps are inoperative due to loss of normal electric power sources or the main feed pumps are inoperative. The AFW system may also be used for normal system cooldown to 300<sup>o</sup> F. By adjusting the feedwater flow to the permissible cooldown rate of 100<sup>o</sup> F per hour, decay heat removal and cooldown of both units can be accomplished in six hours."

6. Final Safety Analysis Report (FSAR) 14.10 LOSS OF FEEDWATER FLOW INCIDENT, states in part:

"An AFW system is available to provide sufficient feedwater flow to remove residual heat generation from the reactor coolant system following reactor trip from full power. The

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control of feedwater flow and steam generator level is by remote manual control from the Control Room.

The inventory remaining in the steam generators following trip is sufficient to allow 13 minutes for AFW initiation before the units are empty.

The Reactor Protection System (RPS) is adequate to protect the core from thermal damage in the event of a complete loss of feedwater incident. The operator will have at least 13 minutes after the initiation of the incident to restore partial feedwater flow by actuation of the AFW system. Controls, annunciators, and flow meters are available at the main control board to permit the operator to take the necessary action."

7. NRC Inspection Report 50-317/81-04; 50-318/81-04, Item 14 (6) states in part:

"Although this action modified the system flow path, the senior control room operator did not require an operational test to determine if the minimum specified AFW flow requirements were satisfied".

**COMMENTS/CORRECTIVE ACTIONS:**

Subsequently, a test was conducted and the flow was determined to be 480 gpm. The results were reviewed and approved by the Plant Operations and Safety Review Committee (POSRC). As reported in LER 81-02/3L, we have since restored full stroke capability to the AFW flow control valves. Additionally, a memo was sent to our Responsible Design Organization (RDO) specifying how improvements could be made to prevent similar situations.

8. NRC Inspection Report 50-317/81-04; 50-318/81-04, Item 14 (9.c), states in part:

"Auxiliary feedwater flow to No. 11 and 12 Steam Generators (SGs) was found to be degraded on January 10, 1981, during plant heatup. Flow to SG #11 was 130 gpm, flow to SG #12 was 350 gpm (480 gpm total). The cause was a flow restriction introduced pursuant to Maintenance Request MR-0-80-6093, issued on December 15, 1980, and signed off as completed on December 17, 1980. That MR limited the maximum travel of flow control valves 1-CV-4511 and 1-CV-4512 to 75% open by an adjustment (handwheel) not involving physical modification of the valves. The purpose of the flow restriction was to prevent AFW pump runout during automatic AFW start on low SG level,

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a condition which could occur on failure of the valve air supplies (which had not been upgraded as a part of the AFW auto-start modification).

The 75% limit had been evaluated (Bechtel Power Corporation letter of May 30, 1980) as providing 460 gpm total flow to both SGs with one AFW pump running. A Combustion Engineering letter of June 4, 1980, stated that 460 gpm flow at 1000 psia would meet residual heat removal requirements; however, Technical Specification Basis 3/4.7.1.2 specified that each AFW pump is capable of delivering a total feedwater flow of 700 gpm at a total dynamic head of 2490 ft, sufficient to remove decay heat and to reduce reactor coolant temperature below 300 degrees F (when shutdown cooling may be placed into operation)."

COMMENTS/CORRECTIVE ACTIONS:

To update the Technical Specification Basis, Baltimore Gas and Electric Company has recently submitted an amendment to the Technical Specification Basis 3/4.7.1.2 which clarifies the design requirements of the AFW System. As the Technical Specification Bases currently reads it specifies that 700 gpm is sufficient to remove decay heat and reduce the RCS temperature to less than 300<sup>o</sup> F, but is silent on the minimum flow requirement. Our records, including the Bechtel Power Corporation letter of May 30, 1980, and the Combustion Engineering letter of June 4, 1980, provided the basis for our determination that 450 gpm was adequate to remove maximum decay heat and achieve cooldown as required by the FSAR in section 10.2.3.

- NOTE -

You will notice the number 450 gpm and 460 gpm throughout this report, referred to as the design basis AFW flow. Although some earlier reference documents used 460 gpm, our RDO has further refined their calculations and determined 450 gpm was adequate for maximum decay heat removal and cooldown of the RCS to 300<sup>o</sup> F. The calculations have been reviewed and the use of 450 gpm or 460 gpm it relates to this report will not change the conclusions.

Interpretation of the AFW flow requirement was further complicated by terminology and method of expression in the Combustion Engineering letter of May 30, 1980, and the Special Instructions written relative to our Facility Change Request (FCR) package 79-1035. The Combustion Engineering letter used the words "residual heat" which have been determined to mean decay heat and cooldown of the Primary Coolant System to 300<sup>o</sup> F. This point was verified through calculations by our RDO.

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9. The Special Instructions in Supplement 4 to FCR 79-1035, stated in part:

"A flow rate of 460 gpm is sufficient to hold at **HOT STANDBY** and maintain steam generator inventory under the worst case heat loads. Under normal conditions a cooldown resulting in a drop in steam generator pressure will occur."

COMMENTS:

This statement was incomplete in that a flow rate of 460 gpm is sufficient to remove decay heat in the most adverse case and cool the RCS to 300<sup>o</sup> F as required by Section 10.2.3 of the FSAR. Based on the foregoing analyses, it is apparent that no violation of Technical Specification 3.7.1.2, "Auxiliary Feedwater System" occurred, since the AFW pumps were indeed operable.

10. Our Licensee Event Report (LER) 81-02/3L on this subject was not entirely correct since the writer was not aware that cooldown capacity was included in the 460 gpm number. This LER assumed 460 gpm was required to remove decay heat and maintain the **HOT STANDBY** condition, without reference to the capability for cooldown. It also assumed the FSAR Safety Analysis in Section 14.10, "Loss of Feedwater Flow Incident", which addressed decay heat removal was the only design requirement relative to AFW flow.

CORRECTIVE ACTION:

In order to prevent recurrence, the Plant Superintendent will ensure LER's are forwarded to the RDO for review when the basis for design or the Safety Analysis is referenced. This action will ensure that adequate and timely communications take place which will in turn ensure that complete and accurate information is provided in reports of this nature.

11. NRC Inspection Report 50-317/81-04; 50-318/81-04, Item 14 states in part:

"The restriction of the AFW Regulating Valve full open position has been performed under Supplement 5 (approved July 9, 1980) to FCR 79-1035. A 10 CFR 50.59 Safety Analysis had been performed for this FCR Supplement. The analysis was dated June 12, 1980, checked June 16, 1980, and reviewed by the POSRC on July 9, 1980, and the Offsite Safety Review Committee (OSSRC) in Meeting 80-16. The Safety Analysis addressed both the addition of single failure proof time delayed automatic start of AFW flow to the SGs and reduction of initial flow by using the AFW Regulating Valve handwheels. The summary statement concluded that the handwheel

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restriction allowed for sufficient AFW flow to maintain hot standby with worst case decay heat load. The section addressing the margin of safety as defined in the Technical Specification Basis was not completed. The licensee stated that this omission was an oversight, however, they did not believe the T.S. basis required that the stated flow be available solely from the Control Room, and that the system could have achieved the basis flows by local manual operation."

**COMMENTS/CORRECTIVE ACTION:**

Regarding the incomplete Safety Analysis, our RDO has been informed of the need to complete each section of the Safety Analysis. However, the margin of safety was not reduced and our contention that design flows could be achieved locally becomes a moot point since the design flow was available solely from the Control Room (i.e., 450 gpm).

Other specific items in your inspection report addressing procedures and training as they relate to this incident become moot if the proper design information and proper interpretation thereof are communicated more effectively and in a more timely manner to site personnel. The review of selected LERs by our RDO, as previously noted, will quickly bring matters such as these to a satisfactory resolution.

12. NRC Inspection Report 50-317/81-04; 50-318/81-04, Item 14 states in part:

"In summary, the inspector concluded that a change had been instituted which reduced AFW System flow, as controlled from the Control Room, below the T.S. basis value. The change had been initiated without adequately implementing procedural requirements and training to ensure flow was restored locally to the basis value, if required, and no application was made for license change to the NRC."

**COMMENTS:**

In response to this item of apparent noncompliance and the inspection summary we conclude, the modification of the AFW System did not reduce flow, as controlled from the Control Room, below the required value. The design basis flow was always available and corrective actions have been taken or will be taken as noted in this reply to preclude noted communications and implementation inadequacies from occurring in similar situations. Since the actual design bases flow was not reduced, neither was the margin of safety reduced, and no application for a license change was necessary.