BALTIMORE GAS AND ELECTRIC COMPANY

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May 21, 1980

ARTHUR E. LUNDVALL, JR.
VICE PRESIDENT
SUPPLY

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

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

Subject: Calvert Cliffs Nuclear Power Plant
Units Nos. 1 & 2, Dockets Nos. 50-317 & 50-318
Automatic Initiation of Auxiliary Feedwater System

References: a) NRC IE Bulletin No. 80-04 dated 2/8/80,
b) BG&E letter dated 1/25/80 from A. E. Lundvall, Jr. to R. W. Reid, Chief, same subject,
c) BG&E letter dated 2/12/80 from A. E. Lundvall, Jr. to B. H. Grier, same subject, and
d) NRC letter dated 3/27/80 from D. G. Eisenhut to A. E. Lundvall, Jr., Automatic Initiation of AFWS Flow

Gentlemen:

Reference (a) requested that we perform analyses to determine the potential for containment overpressure or return-to-power following a main steam line break accident inside containment with continued feedwater flow. Reference (b) provided, as a result of a separate NRC request, an evaluation of the potential for such an occurrence con erning the use of auxiliary feedwater. Reference (c) transmitted this evaluation to you as our response to Reference (a).

As a result of our continuing evaluation of the auxiliary feedwater system to comply with the requirements of NUREG 0645 and NUREG 0578, we are forwarding this supplement to Reference (c) which provides additional information, including a proposed action concerning auxiliary feedwater discharge valves, for a further assessment of the auxiliary feedwater system. This assessment was scheduled in Reference (d) for completion by June 1, 1980. We feel that this proposed action and the analysis submitted previously consititute a sufficient response to the concerns addressed in Reference (a).

Very truly yours,

Arthur E. Lundvall, Jr.

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cc: J. A. Biddison, Esquire
G. F. Trowbridge, Esquire
Mr. E. L. Conner, Jr.

RESPONSE TO IE BULLETIN 80-04

1. Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow.

   Response

   The containment pressure response analysis described in reference (a) accounted for auxiliary feedwater (AFW) flow at the runout flow rate of 2200 gpm. The entire AFW flow, which was assumed to reach the ruptured steam generator, was assumed to be initiated 180 seconds after the main steam line break. In addition, main feedwater was assumed to ramp down to 5% of full power feedwater flow in 60 seconds, and remain at that rate throughout the transient. A more realistic main feedwater flow would ramp down to zero in 20 seconds. This assumption produces a more severe reactivity transient.

   The ability to detect and isolate the damaged steam generator was not considered. This provides the maximum potential water inventory to the containment for calculating the maximum containment pressure and to verify that the facility can withstand a main steam line break accident without identifying the ruptured steam generator. The auxiliary feedwater (AFW) pumps cannot run without severe cavitation at the runout flow used in the analyses described in Reference (a). With two pumps operating, against maximum steam generator pressure, a maximum combined flow rate of 960 gpm can be reached without cavitation to either pump. Therefore, in order to protect the auxiliary feedwater pumps, administrative controls shall be initiated on June 1, 1980 (pending NRC approval) to set and to maintain the AFW pump discharge valve at a 57% open position. This setting will prevent cavitation of the pump at all pressures above the steam generator isolation signal setpoint, while providing adequate flow to the steam generators in order to maintain an adequate heat sink.

   The 960 gpm maximum flow rate for the AFW pumps would result in lower peak containment pressure than that in Reference (a).

2. Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

a. The boundary conditions for the analysis e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.

b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

c. The effect of extended water supply to the affected steam generator on the core criticality and return to power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Radio (MDNBR) values for the analyzed transient.

Response

The analyses described in Reference (a) were performed assuming that the most reactive control rod was in the fully withdrawn position, and water sources were as listed above. Pertinent analytical details are described below.

a. Moderator temperature coefficient used is described by a curve of reactivity insertion versus temperature, which was presented as Figure 1 of Reference (a). Both analyses in Reference (a) used a 2754 MWt power level for the full power analyses and 1 MWt for the zero-power analyses. The end of life shutdown margin was assumed to be 6.4% $\Delta\rho$. Low steam generator level was assumed to occur at the outset of each analysis.

b. Each analysis in Reference (a) assumed the operation of only one high pressure and one low pressure safety injection pump. A conservatively low value of -1.0 $\Delta\rho$ per 95 PPM for boron reactivity worth was also assumed.

c. Reference (a) presents the return-to-power analysis assuming a main steam line break with 2200 gpm auxiliary feedwater flow plus 5% main feedwater flow feeding the ruptured steam generator.

d. Hot channel factors and a minimum DNBR ratio value are not applicable for this transient since the critical heat flux was not approached during the transient. This statement is also made in Reference (a).

3. If the potential for containment overpressure exists or the reactor-return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed.

Response

As discussed in Reference (a), the reactor return-to-power and containment overpressure analyses demonstrate that the reactor does not return to power and there is no containment overpressure condition as a result of main steam line break with auxiliary feedwater flow initiated after a minimum three minute time delay. Automatic initiation of auxiliary feedwater will be incorporated with the associated time delay after NRC approval for this modification is obtained. This approval, described in Reference (b), is expected by June 1, 1980. Until that installation is complete, auxiliary feedwater will be manually controlled, and administrative controls have been initiated to prevent operation of the AFW system during the first several minutes following a reactor trip. The runout flow rate of 2200 gpm used in Reference (a) did not take into account the NPSH considerations associated with AFW flow.