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SUBJECT: Responds to 820121 & 0316 requests for addl info re IE Bulletin 80-04, "Analysis of PWR Main Steam Line Break w/ Continued Feedwater Addition." Tasks to identify affected steam generator & isolate auxiliary feedwater flow listed.

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SEP 03 1982

Office of Nuclear Reactor Regulation
ATTN: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
United States Nuclear Regulatory Commission
Washington, D.C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
ADDITIONAL INFORMATION - MAIN STEAMLINE BREAK
WITH CONTINUED FEEDWATER ADDITION

Dear Mr. Varga,

This letter is in response to your letters, dated January 21, and March 16, 1982 in which you requested additional information regarding our response to IE Bulletin 80-04, "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition." Each of your questions is listed below, followed by our response.

NRC Item 1. Please provide the following information concerning your analysis of containment pressure response to a MSLB with continued feedwater addition:

Question 1: A determination of runout AFW flow to the affected steam generator. This should be determined from the manufacturer's pump curves at zero backpressure, unless the system contains reliable anti-runout provisions or an actual backpressure value has been conservatively calculated.

CP&L Response: In the original analysis, the back pressure value was conservatively calculated for the AFW pumps in the runout condition. The back pressure value calculation considered elevation differences and line resistance between the AFW pump's outlet and the steam generator inlet. The steam generator pressure was assumed to be atmospheric, i.e., zero gage. The runout flow at this pressure for each motor driven AFW pump is 316 gpm, as calculated in the original analysis.

Question 2: An evaluation of the potential for a single active failure in the MFW system which could cause the greatest feedwater flow to the affected steam generator during a MSLB accident and a determination of MFW flow rate to the affected generator if a single active failure were to occur.

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CP&L Response: For this event the Main Feed Water Pumps trip, Feedwater Control Valves and Block Valves close. Therefore, no single active failure in the MFW system prevents termination of flow to the affected steam generator during a MSLB accident.

Question 3: If your response to requests 1 and 2 above, change your response to IE Bulletin 80-04, dated May 9, 1980, provide an evaluation of the potential for exceeding containment design pressure using the feedwater runout flow rates identified in Item 1, Requests 1 and 2, above.

CP&L Response: The responses in 1 and 2 above do not change the CP&L response to IE Bulletin 80-04, dated May 9, 1980.

Question 4: Provide the time after the start of a MSLB that containment design pressure will be exceeded if no operator action is taken to terminate the accident. Provide, also, the magnitude of the peak pressure and the time at which the peak occurs.

CP&L Response: The Design Basis for the original calculation of this event assumes 100 seconds of auxiliary feedwater addition. However, for the analysis described in this response the time is conservatively assumed to be 10 minutes for the operator to take the appropriate action to terminate the accident. Our evaluation of Item 2, question 1 has verified that 10 minutes is a conservative period of time. To assume no operator action to terminate the accident is unrealistic.

Question 5: Provide the tasks for the operator to identify the affected steam generator and isolate the AFW flow to that generator and justification that this can be done in 10 minutes.

CP&L Response: The operator tasks required to identify the affected steam generator and isolate the AFW flow are taken from Emergency Instruction (EI)-1 Appendix B and are as follows:

1. Verify that steamline isolation has occured. If not, manually initiate steamline isolation.

2. Verify the steam dump valves and atmospheric relief valves are closed to insure that the emergency has not resulted from an inadvertant opening of these valves.

3. If the reactor coolant pressure drops below 1300 psig, trip all reactor coolant pumps after safety injection pump operation is verified.

4. Determine if one steam generator has blowdown by observation of steam pressure and isolate the auxiliary feedwater flow to that steam generator.

The plant operations staff has evaluated the time required to respond to this accident and has determined that a trained operator responds in 2-3 minutes. The simulator training staff has also evaluated the response time for this event and has determined that a typical operator trainee response time is 2-5 minutes.

NRC Item 2: Please provide the following information concerning your analysis of reactivity response which results from a MSLB with continued feedwater addition:

Question 1: Provide the longest time for the delay to inject boron taking into account a single active failure.

Verify that time in core life which produces the most limiting moderator temperature coefficient for the MSLB accident was used in your analysis.

Note: A statement that the assumptions of SRP15.1.5 are not considered part of the licensing basis will not be considered responsive to this request.

CP&L Response: The first thing that should be noted is the worst MSLB (for both the containment and the core) occurs for a break between the steam generator and the flow restrictor inside containment, not outside containment. ENC report XN-75-14 contained a typographical error as to break location.

One high head safety injection pump is assumed to fail, which will delay the injection of boron into the reactor. For the current analysis based on low T_{ave}, as performed for Cycle 9 in XN-NF-82-18, the safety injection signal occurs at 10 seconds based on low pressure in the pressurizer; and the boron reaches the core in 43 seconds. A bounding end-of-cycle moderator temperature coefficient curve was used for the analyses. All previous analyses have used bounding end-of-cycle moderator temperature coefficients also. This should provide the additional clarification needed for Item 2.

Question 2: If your response to request 1 and 2 of Item 1 and request 1 of item 2 changes your response to IE Bulletin 80-04 dated May 8, 1980, provide an analysis of the core reactivity response to a MSLB considering the Item 1, Requests 1 and 2 and Item 2, Request 1. Provide justificaiton for your assumptions.

CP&L Response: Our previous response has not changed.

In a separate letter to NRC, dated July 23, 1982, CP&L commited to provide a more detailed analysis of the loss of normal feedwater transient and additional information justifying the adequacy and conservatism of the model used in the steam line break analysis, both of which will provide further analysis and justification for the MSLB with continued feedwater addition accident.

If you have any questions regarding this issue, please feel free to contact my staff.

Yours very truly,

SR Zimmerman

S. R. Zimmerman
Manager
Licensing & Permits

DCS/ce (4025C5T2)

cc: Mr. J. P. O'Reilly (NRC-RII)
Mr. G. Requa (NRC)
Mr. Steve Weise (NRC-HBR)