VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

Central File

May 8, 1980

Mr. James P. O'Reilly
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Serial Number 390
NO/HSM/sjl
Docket Nos. 50-280
50-281
50-338
50-339
License Nos. DPR-32
DPR-37
NPF-4
NPF-7

Dear Mr. O'Reilly:

SUBJECT: I. E. BULLETIN 80-04
ANALYSIS OF A PWR MAIN STEAM LINE BREAK
WITH CONTINUED FEEDWATER ADDITION

This in in response to I. E. Bulletin No. 80-04, "Analysis of A PWR Main Steam Line Break with Continued Feedwater Addition". Our responses for Surry Power Station Unit Nos. 1 and 2 and North Anna Power Station Unit Nos. 1 and 2 are attached.

Very truly yours,

BR Sylvia

B. R. Sylvia
Manager - Nuclear Operations
and Maintenance

Attachment

cc: Director, Office of Inspection and Enforcement
Division of Reactor Operations Inspection
Washington, D. C. 20555

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NORTH ANNA POWER STATION UNIT NOS. 1 AND 2
RESPONSE TO I.E. BULLETIN 80-04
ANALYSIS OF A PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

1. Review the containment pressure response analysis to determine if the potential for containment overpressure exists for a main steam line break inside containment including the impact of runout flow from the auxiliary feedwater system and 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 for the Main Steam Line Break (MSLB) accident is discussed in Section 6.2.1.3.1 of the North Anna FASR. The impact of runout flow from the auxiliary feedwater is included in that analysis. The potential for containment overpressure does not result from this analysis.

NORTH ANNA POWER STATION UNIT NOS. 1 AND 2
RESPONSE TO I.E. BULLETIN 80-04
ANALYSIS OF A PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

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cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position.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 Ratio (MDNBR) values for the analyzed transient.

RESPONSE

The current analyses of main steam line break for North Anna have been reviewed, and it has been concluded that the analysis assumptions are appropriate and conservative. Full main feedwater is assumed in the analysis from the beginning of the transient until isolation at a very conservatively cold temperature. No single failure can result in loss of main feedwater isolation capability following a main steam line break.

In addition to full main feedwater flow, the analysis assumes full rated auxiliary feedwater flow from all auxiliary feedwater pumps throughout the transient. The main steam line break results, which are dominated by steam flow contributions to primary-secondary heat transfer, are not sensitive to assumptions concerning auxiliary feedwater addition. A conservative bounding calculation was performed to evaluate the effect of assuming runout auxiliary feedwater flow as opposed to rated auxiliary feedwater flow in the analysis. The results showed a negligible change in peak core power (less than one (1) percent of the rated value), which will have an insignificant impact on calculated core thermal margins.

It is concluded from this evaluation and in light of the large conservatisms inherent in other analysis assumptions (e.g., available shutdown margin) that the current main steam line break analyses for North Anna are conservative and that the effect of runout feedwater flow discussed above has no impact on the safety conclusions of the FSAR regarding the MSLB accident.

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

No corrective action is required as the potential for containment overpressure does not exist and the reactor return to power response is not worsened as a result of this review.

SURRY POWER STATION UNIT NOS. 1 AND 2
RESPONSE TO I.E. BULLETIN 80-04
ANALYSIS OF A PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

1. Review the containment pressure response analysis to determine if the potential for containment overpressure exists for a main steam line break inside containment including the impact of runout flow from the auxiliary feedwater system and 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

As the Surry 1 and 2 FSAR does not consider the containment pressure response from a main steam line break inside the containment, a comparative analysis was performed with Beaver Valley Unit No. 1 to determine if that analysis could be applied to Surry 1 and 2. From this comparison, it was determined that the only parameter which was not comparable was the auxiliary feedwater runout flow to the affected steam generator. In order to make the Beaver Valley No. 1 analysis applicable, the runout flows will be reduced to a comparable level by installing a flow restricting orifice in each auxiliary feedwater line. With the addition of these orifices, the Beaver Valley analysis may be applied to Surry 1 and 2 and the results of this analysis show that the potential for containment overpressure for a main steam line break inside containment, including continued auxiliary feedwater addition does not exist.

SURRY POWER STATION UNIT NOS. 1 AND 2
RESPONSE TO I.E. BULLETIN 80-04
ANALYSIS OF A PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

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 Ratio (MDNBR) values for the analyzed transient.

RESPONSE

The current analyses of main steam line break for Surry have been reviewed, and it has been concluded that the analysis assumptions are appropriate and conservative. Full main feedwater is assumed in the analysis from the beginning of the transient until isolation at a very conservatively cold temperature. No single failure can result in loss of main feedwater isolation capability following a main steam line break.

In addition to full main feedwater flow, the analysis assumes full rated auxiliary feedwater flow from all auxiliary feedwater pumps throughout the transient. The main steam line break results, which are dominated by steam flow contributions to primary-secondary heat transfer, are not sensitive to assumptions concerning auxiliary feedwater addition. A conservative bounding calculation was performed to evaluate the effect of assuming runout auxiliary feedwater flow as opposed to rated auxiliary feedwater flow in the analysis. The results showed a negligible change in peak core power (less than one (1) percent of the rated value), which will have an insignificant impact on calculated core thermal margins.

It is concluded from this evaluation and in light of the large conservatisms inherent in other analysis assumptions (e.g., available shutdown margin) that the current main steam line break analyses for Surry are conservative and that the effect of runout feedwater flow discussed above has no impact on the safety conclusions of the FSAR regarding the MSLB accident.

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 stated in the responses to items 1 and 2, it is proposed that flow restricting orifices be added to the auxiliary feedwater lines. This modification will be completed during the next scheduled refueling outage. In the interim, the Surry 1 and 2 operating procedures for a steam line break, which presently direct the operators to isolate the affected steam generator after the detection of a steam line break, will be reviewed and strengthened if necessary, and additional simulator refresher training will be performed to ensure that the operators are familiar with the procedures and cognizant of any potential procedure changes.