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April 30, 1980

Mr. Boyce H. Grier, Director
U.S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region I
651 Park Avenue
King of Prussia, Pennsylvania 19406

Subject: Response to IE Bulletin No. 80-04

Dear Mr. Grier:

Enclosed is our response to IE Bulletin No. 80-04. This bulletin was received on February 8, 1980.

Sincerely,

L D White Jr.

L. D. White, Jr.

XC: Office of Inspection and Enforcement
Division of Reactor Operations Inspection
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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Response to IE Bulletin 80-04

Request 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 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 1. Although the Ginna post-steamline break containment pressure analysis in the FSAR did not include the effects of auxiliary feedwater flow to the affected steam generator, it is important to recognize that the evaluation also did not include the benefits of passive and active heat sinks inside containment. Continued feedwater/condensate addition to the steam generator will not occur, since the safety injection signal (generated by a variety of process parameters, including high steam line flow, high containment pressure, and low pressurizer pressure) will close the feedwater control valves and stop the feedwater pumps. The addition of maximum auxiliary feedwater flow to the broken steam generator will eventually require operator action to 1) realign flow to the intact generator, 2) terminate auxiliary feedwater flow to the broken generators. Positive information is available to the operator to determine which is the affected steam generator. Steam generator level instrumentation is located inside containment and steam generator pressure is located outside containment where it would not be affected by the accident environment inside containment. It is expected that, through proper training and by use of the emergency procedures, the operators will be capable of quickly recognizing the steam line break, and will perform the proper operations.

There is substantial time available for the operator to perform the two safety functions noted above. The SEP Safe Shutdown review concluded following their site visit in June 1978 that one steam generator would not boil dry for over thirty minutes. Thus there is substantial time to align flow to the intact steam generator. The termination of auxiliary feedwater flow to the affected steam generator, under the pessimistic circumstances, would require more rapid action (but still easily within the capability of the operators) to maintain containment

pressure below design pressure. The analysis presented in Attachment 1 concludes that, assuming minimum safeguards for containment cooling, auxiliary feedwater flow would have to be terminated in about 26 minutes. With maximum safeguards, this time would be extended to about 44 minutes.

There is no need to consider the operation of the auxiliary feedwater pumps at runout flow. The turbine-driven pumps are controlled by a governor, and will not exceed about 400 gpm. The motor driven pump flow is controlled by the AFW control valves, which receive an automatic throttle signal to 200 gpm from their flow controllers. A potential single failure of the flow controller to control flow to 200 gpm is not considered a worst-case single failure in terms of net energy addition to the containment, since the operation of all containment cooling safeguards (vs. the minimum safeguards assumed in this evaluation) would result in a substantial increase in energy removal from containment.

Request 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 2. Westinghouse Electric Corporation performed the original steam break analysis for Ginna as reported in the FSAR and a reanalysis submitted to the NRC in September 1975. Westinghouse has reviewed the assumptions made for main and auxiliary feedwater flow as they apply to licensing basis steam line break transients. Several of the relevant assumptions used in all core transient analyses follow, and are further explained in the Ginna FSAR.

1. The reactor is assumed initially to be at hot shutdown conditions, at the minimum allowable shutdown margin.

2. For the Condition IV breaks, i.e., double-ended rupture of a main steam pipe, full main feedwater is assumed from the beginning of the transient at a very conservative cold temperature.

3. All auxiliary feedwater pumps are initially assumed to be operating, in addition to the main feedwater. The flow is equivalent to the rated flow of all pumps at the steam generator design pressure.

4. Feedwater is assumed to continue at its initial flow rate until feedwater isolation is complete, approximately 10 seconds after the break occurs, while auxiliary feedwater is assumed to continue at its initial flow rate.

5. Main feedwater flow is completely terminated following feedwater isolation.

Based on the manner in which the analysis is performed for Westinghouse plants, the core transient results are very insensitive to auxiliary feedwater flow. The first minute of the transient is dominated entirely by the steam flow contribution to primary-secondary heat transfer, which is the forcing function for both the reactivity and thermal-hydraulic transients in the core. The effect of auxiliary feedwater runout (or failure of runout protection where applicable) is minimal. Greater feedwater flow during the large steamline breaks serves to reduce secondary pressures, accelerating the automatic safeguards actions, i.e. steamline isolation, feedwater isolation and safety injection. The assumptions described above are

therefore appropriate and conservative for the short-term aspect of the steamline break transient.

The auxiliary feedwater flow becomes a dominant factor in determining the duration and magnitude of the steam flow transient during later stages in the transient. However, the limiting portion of the transient occurs during the first minute, both due to higher steam flows inherently present early in the transient and due to the introduction of boron to the core via the safety injection system.

In conclusion, Westinghouse has evaluated the effect of runout auxiliary feedwater flows in the core transient for steamline break, and based on this evaluation, has determined that the assumptions presently made are appropriate for use as a licensing basis. The concerns outlined in the introduction to IE Bulletin 80-04 relative to, 1) limiting core conditions occurring during portions of the transient where auxiliary feedwater flow is a relevant contributor to plant cooldown; and 2) incomplete isolation of main feedwater flow, are not representative of the Westinghouse NSSS designs and associated Balance of Plant requirements.

The most limiting steam line break determined by Westinghouse was analyzed by Exxon Nuclear Co., Inc. and presented in XN-NF-77-40 Supplement 1, "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant," March 1980. This transient occurs at hot zero power with outside power available and the break occurring at the exit of the steam generator. The Exxon analysis does not specifically account for auxiliary feedwater. However, the Steam Generator heat transfer model, using constant heat transfer coefficients, continues to calculate heat transfer from the primary to the secondary side after the broken steam generator has been estimated to be empty. If auxiliary flow was specifically accounted for, its effect would be negligible during the initial portion of the transient and would have minimal effect during later portions of the transient since by the time the broken steam generator empties, the total system reactivity is negative and core power is decreasing. The additional reactivity addition associated with the slight cooldown due to runout flow is more than negated by the boron reactivity inserted by safety injection. Therefore, the severity of the transient is not increased.

Request 3. If the potential for containment overpressure exists or the reactor-return-to-power response worsens, provide a proposed 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 3. Since neither the potential for containment overpressurization nor the reactor-return-to-power response worsens no corrective action is required.

Request 4. Within 90 days of the date of this Bulletin, complete the review and evaluation required by this Bulletin and provide a written response describing your reviews and actions taken in response to each item.

Response 4. This attachment provides the required 90 day response to IE Bulletin No. 80-04.

Attachment 1 - Containment Energy Balance

The purpose of this evaluation is to determine the length of time available to the operator to terminate AFW flow to the broken steam generator following a steam line break inside containment, prior to containment pressure exceeding 60 psig.

Initial Conditions and Assumptions

a. LOCA energy release to containment (includes credit for containment heat sinks): 191.3×10^6 BTU (taken from Table 14.3.4-2 of FSAR). Results in peak containment pressure of 53 psig.

b. Additional energy to reach 60 psig: 16×10^6 BTU (taken from Fig. 14.3.4-3 of FSAR).

c. Minimum safeguards heat removal capability (1 spray pump and 2 fan coolers) = 55×10^3 BTU/sec = 3.3×10^6 BTU/min. (taken from Fig. 14.3.4-9 of the FSAR).

d. Energy input from 600 gpm to the broken steam generator =

$$600 \frac{gal}{min} \times 1180 \frac{BTU}{lbm} \times 62.4 \frac{lbm}{ft^3} \times \frac{1 \ ft^3}{7.48 \ gal} = 5.91 \times 10^6 \frac{BTU}{min}$$

e. Energy released to containment from the initial steam line break accident blowdown = 140×10^6 BTU (taken from Fig. 14.2.5-10 of the FSAR).

Calculations

1. Additional energy which could be absorbed by containment, taking credit for passive and minimum active containment heat sinks following a steam line break. (from a + b - e above).

   $(191.3 + 16) \times 10^6$ BTU - 140×10^6 BTU = 67.3 MBTU

2. Net energy addition to containment following initial steam line break blowdown (from d-c above):

   $(5.91 \times 10^6 \frac{BTU}{min} - 3.3 \times 10^6 \frac{BTU}{min}) = 2.61 \times 10^6 \frac{BTU}{min}$

3. Operator Action Time = $\frac{67.3 \text{ MBTU}}{2.61 \text{ MBTU/min}}$ = 25.8 min