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May 8, 1980

Docket Nos. 50-348 and 50-364
NRC I.E. Bulletin 80-04

Mr. James P. O'Reilly
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N.W.
Suite 3100
Atlanta, Georgia 30303

Dear Mr. O'Reilly:

Alabama Power Company submits the following response to I.E. Bulletin 80-04, Analysis Of A PWR Main Steam Line Break With Continued Feedwater Addition, dated February 8, 1980, for Joseph M. Farley Nuclear Plant Units 1 and 2.

Item 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:

The main steam line break analyses considered the energy associated with the blowdown of the faulted steam generator. In addition, the Farley Nuclear Plant (FNP) analyses included the energy addition due to auxiliary feedwater (AFW) flow to the faulted generator, steam blowdown from the two intact generators, steam addition due to non-isolable steam line volume, and feedwater addition.

The analyses considered uninterrupted AFW flow to the faulted generator for thirty minutes. This flow was conservatively

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calculated assuming that all AFW pumps operate (i.e. no failures) and deliver flow through the AFW flow limiting orifices. The flow limiting orifices limit the AFW pump runout flows and ensures the AFW pumps remain operable at the runout flows.

The FNP main steam isolation valves are designed to stop forward flow from the steam generator, but are not capable of stopping back or reverse flow. Therefore, the steam break analyses include steam blowdown from the two intact generators to the faulted generator. This blowdown is the forward flow from the intact generators, through the 36" cross tie header, back through the steam line to the faulted generator. This blowdown is terminated by steam line isolation on the intact steam lines.

After the two intact steam generators are isolated, the analyses assumed that the non-isolable steam line volume, i.e., the steam piping between the main steam line isolation valve and the turbine, blows down to the faulted generator.

The analyses conservatively considered additional feedwater flow during the transient. The feedwater isolation valves, feedwater control valves, feedwater pump discharge isolation valves, feedwater pumps, and condensate pumps are all closed/tripped automatically following a steam line break. The extra feedwater addition, to the faulted generator, was calculated assuming offsite power available and the most limiting single failure which would maximize feedwater addition.

Review of the steam line break analyses confirmed that all credible energy addition sources have been considered, consistent with the design basis, and factored into the analyses.

The key symptom of a loss of secondary coolant is abnormally low pressure in one or all steam generators. The key symptom may be accompanied by high steam line flow, high steam line differential pressure or steam generator steam flow/feed flow mismatch. Automatic actions for a loss of secondary coolant are: reactor trip, turbine trip, safety injection, containment phase A isolation, main steam line isolation (occurs on high steam flow coincident with low low Tavg or low steam line pressure or containment pressure of 16.2 psig), and containment spray and containment phase B isolation (occurs at containment pressure of 27 psig). The affected steam generator is identified by comparing the steam generator pressures. The affected steam generator will have very low pressure indicated with the non-affected steam generators indicating near normal pressures. When the affected

steam generator has been positively identified, the operator isolates auxiliary feedwater flow to the affected steam generator.

Item 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 reactor is assumed to have an end of life shutdown margin at no load equilibrium xenon conditions, and the most reactive assembly stuck in its fully withdrawn position. A negative moderator coefficient corresponding to the end of life rodded core is assumed with the most reactive rod in the fully withdrawn position.

Minimum capability for injection of high concentration boric acid solution is assumed corresponding to the most restrictive single failure in the safety injection system. The flow delivered corresponds to one charging pump delivering its full flow to the

cold leg header. No credit has been taken for the low concentration boric acid which must be swept from the safety injection lines downstream of the boron injection tank isolation valves prior to the delivery of high concentration boric acid to the reactor coolant loops.

All auxiliary feedwater pumps are initially assumed to be operating, in addition to main feedwater. Main feedwater is assumed to continue at its initial flow rate until feedwater isolation is complete after which auxiliary feedwater is assumed to continue at its initial flow rate. The core transient results are very insensitive to auxiliary feedwater flow. The initial portion of the transient is dominated entirely by the steam flow contribution to the primary-secondary heat transfer, which is the forcing function for both the reactivity and thermal-hydraulic transients in the core.

Power peaking factors correspond to one stuck rod cluster control assembly at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steam line break. The analysis shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position. Thus there is no cladding damage and no release of fission products to the RCS.

The effect of runout auxiliary feedwater flows in the core transient for st line break has been evaluated and determined that the assumptions made are appropriate. The concerns outlined in the subject bulletin 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 applicable to the FNP design and associated balance of plant requirements.

Item 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:

Based on the response to items one and two, no corrective actions are required.

Yours very truly,

F. L. Clayton, Jr.

BDMcK:de

cc: Mr. R. A. Thomas
Mr. G. F. Trowbridge