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DUKE POWER COMPANY

POWER BUILDING

422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

May 7, 1980

TELEPHONE: AREA 704
373-4083

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

Re: RII:JPO
50-269
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IE Bulletin 80-4

Dear Sir:

With regard to your letter of February 8, 1980 which transmitted the subject bulletin, please find attached our response as required by Item 4.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr. By

KRW:scs

Attachment

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DUKE POWER COMPANY
OCONEE NUCLEAR STATION

RESPONSE TO IE BULLETIN 80-04

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 Oconee FSAR containment pressure response analysis for a postulated main steam line break inside containment considered two cases. The pertinent assumptions and system responses in each case are as follows:

Case 1

Following the main steam line break, feedwater was assumed to remain at 100% until the reactor trip occurred. Subsequently, the integrated control system (ICS) was assumed to close the main and startup feedwater control valves. The operator was assumed to take manual control of the feedwater to the affected OTSG and ensure that the main and startup feedwater control valves remained closed. The unaffected OTSG was controlled at the minimum level (two feet) removing the decay heat generated in the core. The containment pressure response analysis calculated a 13 psi rise in the containment pressure, well below the design pressure of 60 psi.

Case 2

Following the main steam line break, the ICS was assumed to close the main and startup feedwater control valves after the reactor trip. No operator action to isolate the affected OTSG was assumed. Following the initial blowdown of the affected steam generator, the main feedwater flow was assumed to be supplied to the affected steam generator at a rate sufficient to maintain the design two foot level. The reactor coolant system was calculated to cooldown to the saturation temperature (284°F) corresponding to the reactor building pressure of 38 psig calculated to occur approximately 250 seconds after the break. Subsequent addition of mass and energy to the containment to remove the core decay heat was insufficient to cause the containment pressure to reach the design building pressure with the reactor building coolers operating.

The existing FSAR analysis summarized above did not explicitly address the impact of the runout flow from the auxiliary feedwater system. This is because continued feedwater addition to the affected steam generator by means of the main feedwater system is considered to be more limiting with respect to containment pressure response than the case involving auxiliary feedwater flow.

The auxiliary feedwater system currently in use at Oconee consists of a turbine driven pump and two motor driven pumps. Each steam generator can receive auxiliary feedwater flow from one motor-driven pump and the turbine driven pump. Operation of the auxiliary feedwater system at runout conditions would result in approximately 2050 gpm (700 gpm from the motor driven pump and 1350 gpm from the turbine driven pump) auxiliary feedwater addition into the affected steam generator. Since the flow capacity of the auxiliary feedwater system is less than that of the main feedwater pump (greater than 10,000 gpm), since the auxiliary feedwater temperature (90°F) is less than that of the main feedwater (460°F) and since the existing analysis considered the maximum possible cooldown of the primary system for a steam line break in the containment, it is concluded that the existing analysis of containment pressure response bounds the situation involving flow from the auxiliary feedwater system.

It is to be noted that continued feedwater addition either by means of the main feedwater system or the auxiliary feedwater system in an uncontrolled manner could ultimately result in overpressurization of the containment. The Oconee Nuclear Station emergency procedure includes the requirement to isolate all feedwater into the affected steam generator. The symptoms indicative of a steam line break and the required manual actions are addressed in this procedure. The operator can isolate the feedwater into the affected steam generator either by closing the flow control valves or the associated block valves. In the event of damage to the turbine driven pump and the motor driven pump due to extended operation at runout flow conditions, the remaining motor driven pump or the main feedwater pump could be utilized for feedwater supply. Additionally, the decay heat removal system could be placed in operation because of the depressurization and cooldown of the primary system resulting from operation of the auxiliary feedwater system at runout conditions for extended periods of time.

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 then 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 Oconee FSAR analysis of the reactivity increase resulting from a main steam line break considered four cases involving various potential modes of feedwater addition to the affected steam generator. In all cases, a minimum rod worth, based on the maximum worth rod considered stuck-out, and consistent with the minimum shutdown margin required by the Technical Specifications was utilized in the core reactivity calculation. The pertinent assumptions and system responses for each of these cases are summarized below.

Case 1

In this case, the integrated control system (ICS) was assumed to initially close the main and startup feedwater control valves following the reactor trip and then the operator was assumed to maintain feedwater isolation of the affected steam generator. The minimum (two foot) level was maintained in the unaffected steam generator. Under these assumptions, the reactor was calculated to remain subcritical throughout the transient.

Case 2

In this case also the ICS was assumed to close the main and startup feedwater control valves following the reactor trip; however, no credit was taken for operator action to maintain feedwater isolation of the affected steam generator. Consequently, feedwater flow by means of the main feedwater pump continued to the affected steam generator at a rate necessary to maintain a two foot level. The resulting cooldown of the primary system was calculated to cause a return to power of about 1% FP at approximately 170 seconds. The core then returned to subcritical conditions with the addition of highly borated water by the emergency core cooling system (HPI, CFT, and LPI).

Case 3

The third steam line break analysis case assumed proper ICS action to initially close the main and startup feedwater control valves; however, no operator action to maintain feedwater isolation of the affected steam generator was assumed. The auxiliary feedwater pump was assumed to start on a low main feedwater pump discharge pressure signal. The ICS was assumed to maintain a minimum (two foot) level in both steam generators with a combination of main and auxiliary feedwater. The analysis predicted a return to 35 percent of rated power in approximately 65 seconds. Without the stuck rod and considering the nominal trip rod worth, the core was found to remain subcritical.

Case 4

The fourth main steam line break analysis case included the assumption of no ICS or operator action to change the feedwater control valve positions. The feedwater flow to the damaged steam generator was postulated to be 135% of the rated flow in one steam generator. It was assumed that the auxiliary feedwater system was not actuated. Under these conditions, the reactor was calculated to return to less than 8 percent of rated power approximately 166 seconds after the break before going subcritical again by injection of borated water by the ECCS.

From the foregoing discussion, it is seen that the existing analysis of the steam line break accident considered several potential modes of feedwater addition to the affected steam generator from the main and auxiliary feedwater systems. Although the flow capacity of the auxiliary feedwater system has increased with the recent addition of the motor driven pump into each of the two secondary loops, the analysis for Case 3 above still represents the worst case core reactivity increase. Since the increases in the auxiliary feedwater flow capacity is very small (less than 5%) compared to the available total feedwater flow capacity and since the amount of feedwater flow into the steam generator is dictated by the steam generator level requirement.

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

As demonstrated in the response to Item 1 above, the potential for containment overpressure is not introduced by postulated auxiliary feedwater pump operation at runout conditions. Furthermore, the existing emergency procedure includes operator guidance to prevent uncontrolled feedwater addition to the affected steam generator. The reactor return-to-power responses calculated in the FSAR still represent the limiting case for core reactivity increase. Therefore, no corrective actions are considered necessary at this time for Oconee Nuclear Station. It is pointed out that a probabilistic risk assessment study is being planned for Oconee. If the results of this study indicate the need for any corrective actions with respect to the steam line break accident, appropriate corrective actions will be considered at that time.