Florida
Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

April 21, 1992
3F0492-10

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Subject: High Energy Line Break (HELB) in the Intermediate Building

References: A. FPC to NRC letter, 3F0690-14, dated June 15, 1990
B. NRC to FPC letter, 3N0690-14, dated June 20, 1990
C. FPC to NRC letter, 3F1289-11, dated December 18, 1989

Dear Sir:

The purpose of this letter is to advise the NRC that Florida Power Corporation (FPC) has completed its evaluation of the environmental conditions created in the Intermediate Building due to a possible terminal end break in either of the steam supply lines to the emergency feedwater pump (EFP) turbine. Reference A discusses the HELB problem. Reference B granted FPC permission to operate CR-3 until the end of Refuel 8 using the fire sprinkler system as interim protection measure. FPC has performed further evaluations and concludes that Crystal River Unit 3 (CR-3) can continue to operate in a safe manner without reliance on the fire sprinkler system as an interim protection measure to maintain compartment temperatures below environmental qualification limits. A December 4, 1991 meeting was held to brief the NRC staff on the progress of the evaluation at that time. This letter will discuss the background, evaluation of the environmental conditions/break locations, relative risk significance, and conclusions.

BACKGROUND

During the completion of the HELB program that was implemented during Refuel 7 in 1990, FPC identified that a postulated terminal end break in either of two 6 inch lines to the EFP turbine may create more harsh environmental conditions in the Intermediate Building than originally considered in establishing the design conditions to assure compliance with 10 CFR 50.49. A break in one of these lines could produce high temperatures on components associated with the main steam and feedwater isolation function of the Emergency Feedwater Initiation and Control (EFIC) System. To assure that CR-3 could continue to operate in compliance with 10 CFR 50.49, FPC began an evaluation of the components that comprise the lines supplying

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steam to the emergency feedwater pump turbines to determine if the location of the terminal end break was postulated realistically. That evaluation and its conclusions are presented in the following discussion.

EVALUATION OF ENVIRONMENTAL CONDITIONS & TERMINAL END BREAK LOCATIONS

A postulated terminal end break in the 6 inch portion of the EFP turbine steam supply piping could produce temperatures in the immediate area which exceed the environmental qualification limits of certain EFIC System electrical components. The 6 inch terminal breaks were identified as breaks AS-1 and AS-5 in the FPC HELB Program Report, "Pipe Rupture Analysis Criteria Outside the Reactor Building," which was last revised by Reference C. As a result of this latest re-evaluation, the postulated terminal end breaks shown as locations AS-1 and AS-5 in the HELB Report have been determined to be more likely to occur at the 3 inch ends of the 6 inch x 3 inch reducers which connect to the inlets of valves MSV-56 and MSV-55, respectively. The temperature created by a postulated break is acceptable and does not violate the qualification temperature limits of the electrical components. A simplified sketch is attached to aid in understanding the configuration of the terminal end piping being evaluated.

The evaluation of the material properties and the section properties of these terminal end components allows FPC to conclude the location having the highest potential for failure is the 3 inch connection from the 6 inch x 3 inch reducer to the 3 inch valve inlet. This location (shown on the simplified sketch) has the smallest diameter and the smallest wall thickness. FPC has determined that the welds at both ends of each reducer were made using the same welding procedures and filler materials. A given axial or shear force on these joints will produce a stress in the 3 inch end that is 2.5 times greater than the 6 inch end. Likewise, a given moment applied to these joints will produce a stress in the 3 inch weld that is 4.8 times greater than in the 6 inch end. Tensile test data for the reducer and the weldolet were reviewed to assure that the materials in the 3 inch joint were not significantly stronger than 6 inch joint. Pipe stress calculations for these piping configurations have also confirmed these conclusions. In each case, the highest break/crack location stress for the entire line from main steam header to EFP turbine is at the 3 inch connection to the inlet of the angle valves.

In support of this effort to define terminal end rupture locations, FPC performed updated pipe stress analyses of the two steam lines to the EFP turbines. The analysis for the line containing MSV-55 did show an increase in stress levels at two locations which would be above the crack postulation limits specified in Section 4.1.7 of the HELB Program Report. These new cracks are located at the 3 inch connections to the inlet and outlet of MSV-55. The environmental effects of these new cracks have been evaluated and found to be bounded by the conditions created by previously evaluated cracks in these areas. The analysis for the line containing MSV-56 did show piping stress levels above B31.1 code allowable; but piping structural integrity would be maintained, and, therefore, the system would not fail and would remain operable. A new piping restraint is being added to this line during Refuel 8 to reduce the piping stress levels to less than code allowables.

RELATIVE RISK SIGNIFICANCE

FPC has also evaluated the potential impact of this postulated break using the CR-3 PRA model. NUREG/CR-4407, "Pipe Break Frequency Estimation for Nuclear Power Plants," gives the frequency of such a break as 5.0×10^{-4} per year. The postulated pipe break would render the steam-driven emergency feedwater pump inoperable. The increased steam demand would not cause CR-3 to experience a reactor trip. However, FPC conservatively postulated a reactor trip. Using concurrent reactor trip and pipe break as an initiating event in the CR-3 model, no core damage sequences with a frequency greater than or equal to 1×10^{-8} per year were found. As a comparison, the total CR-3 core damage frequency for internal events is 1.4×10^{-5} per year.

CONCLUSIONS

FPC can conclude from the analyses performed on these steam supply lines to the EFP turbine that the greatest potential for a postulated terminal end rupture will occur at the 3 inch end of the 6 inch x 3 inch reducer. If a rupture were postulated in the 3 inch terminal end, the environmental effects on surrounding electrical equipment subject to the requirements of 10 CFR 50.49 will remain less harsh than the conditions for which they are qualified. The pipe whip and jet impingement interactions created by postulated 3 inch ruptures will be bounded by the interactions previously evaluated for the postulated 6 inch terminal end breaks at the connection of the weldolets to the reducers. The pipe whip restraints and jet shields in these areas were designed for the larger postulated breaks. The new postulated cracks will also remain bounded by previous analyses. Based upon the CR-3 PRA model analysis, this postulated pipe break is judged not to be risk-significant.

In the December 4, 1991 NRC/FPC meeting, FPC stated that it might be necessary to perform additional examinations and fracture mechanics analyses to support the postulation of the breaks at the 3 inch end of the reducers. Following the review of the material and section properties of the reducers and the results of the pipe stress analyses, FPC has concluded that these additional examinations and analyses are not necessary. There is a reasonable assurance that the postulated breaks will occur in the smaller 3 inch end of the reducer rather than the larger 6 inch end.

The results of this program lead FPC to conclude that CR-3 does not need to rely on the fire sprinkler system as interim protection for the adjacent equipment. CR-3 is in compliance with its HELB Program and 10 CFR 50.49.

FPC is also submitting as an attachment six (6) copies of the replacement pages for the FPC HELB Program Report mentioned above. The report is being revised to clarify terminal end break locations AS-1 and AS-5 and to identify two new postulated crack locations at nodes 100 and 103 of Auxiliary Steam Problem CR-4A.

Sincerely,

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

PMB/JWT

Attachments

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

SIMPLIFIED SKETCH OF MSV-55 AND MSV-56 CONNECTION TO 24 IN MAIN STEAM LINE

Attachment to 3F0492-10