UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION REPORT

MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

FORT CALHOUN STATION, UNIT 1

Docket No. 50-285

1.0 Introduction

In the summer of 1979, a pressurized water reactor (PWR) licensee submitted a report to the NRC that identified a deficiency in its original analysis of the containment pressurization resulting from a postulated main steam line break (MSLB). A reanalysis of the containment pressure response following a MSLB was performed, and it was determined that, if the auxiliary feedwater (AFW) system continued to supply feedwater at runout conditions to the steam generator that had experienced the steam line break, the containment design pressure would be exceeded in approximately 10 minutes. In other words, the long-term blowdown of the water supplied by the AFW system had not been considered in the earlier analysis.

On October 1, 1979, the foregoing information was provided to all holders of operating licenses and construction permits in IE Information Notice 79-24 [2]. Another licensee performed an accident analysis review pursuant to the information furnished in the above cited notice and discovered that, with offsite electrical power available, the condensate pumps would feed the affected steam generator at an excessive rate. This excessive feed had not been considered in its analysis of the postulated MSLB accident.

A third licensee informed the NRC of an error in the MSLB analysis for their plant. For a zero or low power condition at the end of core life, the licensee identified an incorrect postulation that the startup feedwater control valves would remain positioned "as is" during the transient. In reality, the startup feedwater control valves will ramp to 80% full open due to an override signal resulting from the low steam generator pressure reactor trip signal. Reanalysis of the events showed that the rate of feedwater addition to the affected steam generator associated with the opening of the startup valve would cause a rapid reactor cooldown and resultant reactor-return-to-power response, a condition which is beyond the plant's design basis.

Following the identification of these deficiencies in the original MSLB accident analysis, the NRC issued IE Bulletin 80-04 on February 8, 1980. This bulletin required all licensees of PWRs and certain near-term PWR operating license applicants to do the following:

"1. Review the containment pressure response analysis to determine if the potential for containment overpressure for MSLB 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.

"2. Review your analysis of the reactivity increase which results from a MSLB 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; and
d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn positions at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient.

"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."

Following the licensee's initial response to IE Bulletin 80-04, a request for additional information was developed to obtain all the information necessary to evaluate the licensee's analysis. The results of our evaluation for Fort Calhoun Station, Unit 1 (Fort Calhoun 1) are provided below.

2.0 Evaluation

Our consultant, the Franklin Research Center (FRC), has reviewed the submittals made by the Omaha Public Power District in response to IE Bulletin 80-04, and prepared the attached Technical Evaluation Report. We have reviewed this evaluation and concur in its bases and findings.

3.0 Conclusion

Based on our review of the enclosed Technical Evaluation Report, the following conclusions are made regarding the postulated MSLB with continued feedwater addition for Fort Calhoun 1:

1. There is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition

because the main feedwater system is isolated and the auxiliary feedwater actuation system prevents actuation of the affected steam generator from being fed;

2. The AFW pumps will not experience runout conditions; therefore, they will be able to carry out their intended function without incurring damage during a MSLB;
3. All potential water sources were identified and, although a reactor return-to-power due to decay heat and subcritical multiplication occurs, the reactor remains subcritical, and the DNBR remains greater than 1.30; therefore, the Reference 3 reactivity increase analysis remains valid;
4. No further action is required of the licensee regarding IE Bulletin 80-04; and
5. Compliance of the Main Steam Isolation Signal, Containment Isolation Actuation Signal and the Safety Injection Actuation Signal with IEEE Standard 279-1971 was not considered in this review.

4.0 References

1. "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition," NRC Office of Inspection and Enforcement, February 8, 1980, IE Bulletin 80-04
2. "Overpressurization of the Containment of a PWR Plant after a Main Steam Line Break," NRC Office of Inspection and Enforcement, October 1, 1979, IE Information Notice 79-24
3. W. C. Jones (OPPD) letter to K. V. Seyfrit (NRC, Region IV), Subject: IE Bulletin 80-04, Response to Item 2, May 15, 1980

4. W. C. Jones (OPPD), letter to K. V. Seyfrit (NRC, Region IV), Subject: IE Bulletin 80-04, Response to Item, August 27, 1980
5. W. C. Jones (OPPD) letter to H. R. Denton (NRC), Subject: Application for Amendment of Operating License, November 17, 1981
6. Fort Calhoun Station, Unit 1 Final Safety Analysis Report, through Rev. 27, Omaha Public Power District
7. "PWR Main Steam Line Break with Continued Feedwater Addition - Review of Acceptance Criteria," Franklin Research Center, November 17, 1981, TER-C5506-119
8. "Criteria for Protection Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, New York, NY, 1971, IEEE Std 279-1971
9. Standard Review Plan, Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)", NRC, July 1981, NUREG-0800
10. "Criteria for Accident Monitoring Functions in Light-Water Cooled Reactors," American Nuclear Society, Hinsdale, IL, December 1980, ANS/ANSI-4.5-1980
11. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Rev., NRC, December 1980, Regulatory Guide 1.97
12. "Single Failure Criteria for PWR Fluid Systems," American Nuclear Society, Hinsdale, IL, June 1976, ANS-51.7/658-1976
13. "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Rev. 3, NRC, February 1976, Regulatory Guide 1.26

14. "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Rev. 1, NRC, July 1981, NUREG-0588

15. "Auxiliary Feedwater System Automatic Initiation and Flow Indication, Fort Calhoun Station," December 8, 1981, TER-C5257-297

16. R. A. Clark (NRC) letter to W. C. Jones (OPPD), Subject: Auxiliary Feedwater System, February 20, 1981.

Attachment:
Franklin Research Center
Technical Evaluation Report