SAFETY EVALUATION REPORT
CONTAINMENT SYSTEMS BRANCH
MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION
SEQUOYAH NUCLEAR PLANT
UNIT 1

Docket No.: 50-327

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 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 the analysis of the postulated MSLB accident.

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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 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 in the event of a 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 the ability to detect and isolate the damage 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 previcus 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 overpressurization 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 Sequoyah Nuclear Plant, Unit 1, (Sequoyah 1) are provided below.

2.0 Evaluation

Our consultant, the Franklin Research Center (FRC), has reviewed the submittals made by the licensee 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 Sequoyah 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 auxiliary feedwater flow to the affected steam generator is restricted.

2. All potential water sources were identified. Although a reactor return-to-power is predicted there is no violation of the specified acceptable fuel design limits. Therefore, the Final Safety Analysis Report reactivity increase analysis remains valid.

3. The auxiliary feedwater (AFW) pumps are individually protected against the effects of runout flow. A single failure of the runout control system will only a~~e~~ffect one pump, leaving the other two pumps capable of continued operation.

4. No further action regarding IE Bulletin 80-04 is required.

11. "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors"
American Nuclear Society, Hinsdale, IL, December 1980
ANS/ANSI-4.5-1980

12. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,"
Revision 2, NRC, December 1980, Regulatory Guide 1.97

13. "Single Failure Criteria for PWR Fluid Systems,"
American Nuclear Society, Hinsdale, IL, June 1976,
ANS-51.7/N658-1976

14. "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"
Revision 3, NRC, February 1976, Regulatory Guide 1.26

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

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. L. M. Mills (TVA)
Letter to J. P. O'Reilly (NRC, Region II)
Subject: Office of Inspetion aqnd Enforcement Bulletin 80-04-RII: JPO 50-327 - Sequoyah Nuclear Plant Unit 1
June 16, 1980

4. L. M. Mills (TVA)
Letter to E. Adersam (NRR)
Subject: Additional Information on Main Steam Line Break
May 11, 1982

5. Sequoyah Nuclear Plant Unit 1
Final Safety Analysis Report, through Rev. 37
Tennessee Valley Authority, October 1975

6. Watts-Bar Nuclear Plant
Final Safety Analysis Report, through Rev. 39
Tennessee Valley Authority

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, 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