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

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING LARGE BREAK LOCA ANALYSIS ERROR CORRECTION

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-364

1.0 INTRODUCTION

By letter dated February 18, 1991,* Alabama Power Company notified the Nuclear Regulatory Commission (NRC) of an error in the design basis loss-of-coolant accident (LOCA) analysis for Joseph M. Farley Nuclear Plant (Farley), Unit 2. Analyses performed in 1982 using the Westinghouse 1978 large break emergency core cooling system (ECCS) evaluation model had identified that analyses for Farley, Unit 1, whose barrel baffle design had been converted to an upflow configuration, were limiting for Farley, Unit 2, which continued to have a downflow barrel baffle configuration.

Subsequent large break LOCA analyses for Farley, Unit 2, performed using the more recent Westinghouse 1981 large break LOCA evaluation model with BASH (Reference 1) assumed that analyses for the barrel baffle upflow configuration continued to be limiting for Farley, Unit 2. The February 18, 1991, letter reported that the licensee had discovered that this assumption was incorrect, and described an updated analysis amending the barrel baffle flow assumption. The licensee submitted additional clarifying information in a letter dated June 17, 1991.

2. EVALUATION

The NRC staff reviewed the Farley, Unit 2, reanalysis, which included amended input assumptions and the barrel baffle flow correction.

2.1 Large Break Evaluation Methodology

The licensee identified that the reanalysis for Farley, Unit 2, was performed using the Westinghouse 1981 large break evaluation model with BASH. This is an approved methodology, applicable to the Farley, Unit 2, design, and acceptable for this application.

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* Subsequent to this submittal, Amendment Nos. 90 and 83 to Facility Operating Licenses NPF-2 and NPF-8, respectively, were issued authorizing Southern Nuclear Operating Company, Inc., to become the licensed operator. This change was implemented on December 23, 1991.

2.2 Analysis Assumptions

In addition to correcting the barrel baffle flow assumption for the Farley, Unit 2, analysis, the licensee also took credit for injection of the volume of water contained in the accumulator lines between the accumulator tanks and the first check valves from the tanks. A review of the Farley, Unit 2, piping diagrams in the Final Safety Analysis Report indicated that the sections of accumulator piping under consideration may reasonably be considered extensions of the accumulators and, therefore, that credit for this volume of water in the analysis is acceptable.

In previous Farley, Unit 2, LOCA analyses, a zero containment backpressure was conservatively assumed. The updated Farley, Unit 2, LOCA analysis assumes a containment backpressure calculated using the approved Westinghouse COCO computer code (Reference 2). Because the COCO code has been approved for this use, we find the use of the calculated containment backpressure acceptable for the Farley, Unit 2, analysis.

Additional assumptions are as follow:

A. The reactor is fueled with Westinghouse low parasitic (LOPAR) fuel.

B. Peak linear power is 102 percent of 12.314 kw/ft.

C. Peaking factor (at design rating) is 2.32.

D. Steam generator tube plugging is a maximum of 20 percent in any one steam generator with an average of 15 percent for all three steam generators.

Other significant assumptions (e.g., limiting single failure) remain unchanged.

2.3 Analysis Results

Using the assumptions discussed in Section 2.2, the licensee reanalyzed the break identified by previous analyses as the limiting large break LOCA for Farley, Unit 2. This break is a double-ended cold leg guillotine rupture with a discharge coefficient of 0.4. The licensee's reanalysis calculated a peak cladding temperature of 2163° F with a corresponding maximum local oxidation of 8.09 percent and total core-wide oxidation of less than 0.3 percent. These results are within the limits specified in 10 CFR 50.46(b)(1), (2) and (3) of 2200° F, 17 percent, and 1 percent, respectively. The results indicate that the core will remain amenable to cooling, and that the Farley, Unit 2, emergency core cooling design, as approved, provides adequate long-term cooling capability. Therefore, there is assurance that the requirements of 10 CFR 50.46(b)(4) and (5) will continue to be met.

3.0 CONCLUSIONS

Based on our review, t'e NRC staff concludes that the Farley, Unit 2, large break LOCA analysis was performed using acceptable methods and assumptions. The calcuiated results meet the requirements of 10 CFR 50.46(b); and are, therefore, acceptable.

4.0 REFERENCES

1. WCAP-10266-P-A, Revision 2 (Proprietary), WCAP-10267-A, Revision 2 (Non-proprietary), "1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," Besspiata, J.J., et al., March 1987

2. WCAP-8327 (Proprietary), WCAP-8326 (Non-proprietary), "Containment Pressure Analysis Code (COCO)." Bordelon, F.M. and Murphey, E.T., June 1974

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Date: March 31, 1992