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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO OPPD-NA-8303 REV. 03

TRANSIENT AND ACCIDENT METHODS AND VERIFICATION

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By letter dated April 19, 1991, as supplemented by letter dated November 27, 1991, Omaha Public Power District (OPPD) proposed changes to Topical Report OPPD-NA-8303, "Reload Core Analysis Methodology, Transient and Accident Methods and Verification." The report describes OPPD's reload core transient and accident methods for application to Fort Calhoun Station, Unit No. 1. The proposed changes would be incorporated as Revision 3 and would show Westinghouse as the analyst for the large and small break loss of coolant accident (LOCA) evaluations and the control element assembly (CEA) ejection analysis. The excess load and boron dilution events have also been reclassified to obtain operating margin in the departure from nucleate boiling (DNB) limiting safety system settings (LSSS). The proposed revision references the ABB Combustion Engineering Nuclear Transient Simulation (CENTS) code, which would eventually replace the ABB Combustion Engineering System Excursion Code (CESEC-III) for transient thermal hydraulic calculations. Comparisons of results using both nuclear steam supply system (NSSS) simulation codes to results from experimental measurements and independent calculations are presented.

2.0 EVALUATION

The report references the CENTS computer code and discusses OPPD's verification of CENTS in the simulation of plant response to non-LOCA initiating events. However, the NRC generic review of CENTS (Ref. 1) for Combustion Engineering (CE) has not been scheduled specifically for OPPD's Fort Calhoun Station. In the interim, OPPD will continue to use the CESEC-III code to simulate plant transients.

The large and small break LOCA are analyzed for OPPD by Westinghouse (Ref. 2). OPPD confirms that the assumptions used in these analyses are valid for each reload core and that, if reanalysis is required, it will be performed by Westinghouse. The applicability of the Westinghouse LOCA evaluation models to Fort Calhoun is currently under review in a separate licensing action.

The excess load event has been deleted as being one of the transients analyzed to determine the transient response term applied to the Thermal Margin/Low Pressure (TM/LP) trip equation. This deletion is acceptable, since DNB protection from the excess load event will now be achieved by building

sufficient initial margin into the DNB Limiting Conditions for Operation (LCOs) in conjunction with the Variable High Power Trip (VHPT). The transient response term applied to the TM/LP equation in the Technical Specifications will now be a result solely of the analysis of the Reactor Coolant System (RCS) depressurization event.

The analysis assumptions for the boron dilution event initiated from hot standby, hot shutdown, or cold shutdown conditions have been modified by adding a term to correct for density changes due to the temperature difference between the makeup water and the reactor coolant water. This change ensures a conservative operating temperature range and is consistent with CE methodology options previously approved by the NRC for application in reload methodology calculations. Therefore, the change is acceptable.

The analysis of the CEA ejection event will be performed for OPPD by Westinghouse (Ref. 3). The acceptable limiting criteria are as follows:

1. The radial average pellet enthalpy at the hot spot shall not exceed 280 cal/gram.

2. The peak reactor pressure during any part of the transient will be less than the value that will cause stress to exceed the emergency condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code. This objective is achieved if the peak RCS pressure does not exceed 2750 psia (110% of design).

3. Offsite dose consequences will be within the guidelines of 10 CFR Part 100.

4. Offsite dose consequences will be based on the number of fuel rods that fail because of DNBR values that are below the analysis Critical Heat Flux (CHF) correlation limit. An explicit number of rods will not be calculated. Instead, a conservatively bounding value of 10%, based on the approved vendor methodology, will be used.

3.0 CONCLUSION

The staff has reviewed the proposed changes in Revision 3 of OPPD-NA-8303 and finds them acceptable. However, use of the CENTS computer code by OPPD for transient analyses has not been reviewed as a part of this licensing action and is not approved at this time. In the interim, OPPD will continue to use the CESEC-III code to simulate plant transients. The applicability of the Westinghouse LOCA evaluation models to Fort Calhoun is also being evaluated in a separate licensing action.

4.0 REFERENCES

1. Letter from S. A. Toelle (ABB-CE) to T. R. Quay (NRC), LD-91-044, Request for Generic Review of CENTS Code, August 8, 1991.

2. "Westinghouse ECCS Evaluation Model for Analysis of CE-NSSS," WCAP-13027-P, July 1991.

3. "Control Element Assembly Ejection Accident Methodology Summary Report," Omaha Public Power District, Fort Calhoun Unit 1, November 27, 1991.

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