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ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO

OPPD-NA-8303 REV. 04

TRANSIENT AND ACCIDENT METHODS AND VERIFICATION

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT 1

DOCKET NO. 50-285

1.0 INTRODUCTION

In a letter of February 5, 1993, Omaha Public Power District (OPPD) submitted 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 4.

2.0 EVALUATION

The report references the ABB Combustion Engineering (ABB-CE) CENTS computer code and discusses OPPD's verification of CENTS in the simulation of plant response to non-loss-of-coolant accident (LOCA) initiating events. ABB-CE requested generic review of CENTS (Ref. 1), and NRC review and approval of CENTS has been completed (Ref. 2). OPPD will use CENTS in place of the CESEC-III code to simulate plant transients. In order to verify CENTS for Fort Calhoun safety analyses and to demonstrate their ability to correctly use CENTS, OPPD has benchmarked their results against both actual plant transient data and independent safety analyses performed with either CESEC-III or other codes previously accepted by the NRC.

The plant transients which were benchmarked were the turbine-reactor trip and four-pump loss of coolant flow events. Both the CENTS and CESEC-III predicted parameters for the turbine trip show very good agreement with those measured in the Fort Calhoun Cycle 1 startup testing performed at nominal full power conditions. In addition, the CENTS predicted reactor coolant system (RCS) flow coastdown, as well as the CESEC-III predictions for the 35% power total loss of coolant flow, show very good agreement with the parameters measured during Cycle 1 startup testing.

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No plant data existed for the transients analyzed by OPPD for reload core licensing using ABB-CE methodology. Therefore, a comparison of limiting events to previous analyses performed by OPPD using CESEC-III was done. The events chosen for comparison were the dropped control element assembly (CEA) event and the RCS depressurization event. For the dropped CEA comparison, the primary system responses between the CESEC-III and CENTS predictions showed excellent agreement with each other. For the RCS depressurization event, CENTS and CESEC-III gave very similar predictions for core power versus time, except for the time of trip. However, this was due to the fact that the trip in CENTS was based on the actual TM/LP control system, whereas for CESEC-III, a manual trip was simulated at the time of maximum margin degradation. Comparisons of the RCS pressure response between the CENTS and CESEC-III analyses show very good agreement post-trip.

The main steamline break event was not benchmarked using CENTS since the three-dimensional core neutronics model was not available during the benchmark effort. Therefore, CENTS will not be used in a licensing application for analysis of a main steamline break event until a future benchmark is performed.

The large and small break LOCA are analyzed for OPPD by Westinghouse (Ref. 3). OPPD confirms the assumptions used in these analyses are valid for each reload core and, if reanalysis is required, it is performed by Westinghouse. The applicability of the Westinghouse LOCA evaluation models for Fort Calhoun has been approved by the NRC (Ref. 4).

The reference to the CETOP Code for calculating the minimum departure from nucleate boiling ratio (DNBR) in the CEA withdrawal event has been changed to TORC since the TORC Code will now be used for calculating the required overpower margins. This is acceptable since both codes have been approved by the NRC for calculating fuel rod DNBR.

Revision 4 incorporates the OPPD steam generator tube rupture methodology. This methodology was approved by the NRC by safety evaluation dated May 22, 1991 (Ref. 5).

Various other minor revisions have been made to reflect current approved methodology, to remove cycle specific results, to make editorial corrections, and to update references to the most recent revisions. These various changes have been found to be acceptable.

3.0 CONCLUSION

The staff has reviewed the proposed changes in Revision 4 to OPPD-NA-8303 and finds them acceptable. The CENTS code has been previously reviewed generically and found to be acceptable for plant transient analyses. Use of the CENTS computer code by OPPD for transient analyses has been reviewed as a part of this licensing action and is approved subject to the same conditions and limitations specified in the NRC generic approval. Specifically, the CENTS DNBR calculation for determining overall trends in thermal margin should not be used for licensing analyses. Adequate benchmarking of the CENTS LOCA and CEA ejection accident capabilities has not been provided. Consequently, CENTS should not be used for performing LOCA or CEA ejection accident licensing analyses. Benchmarking for the CENTS three-dimensional core neutronics capability has not been provided and, consequently, licensing applications of CENTS must use the point kinetics model. The applicability of the Westinghouse LOCA evaluation model and the Westinghouse CEA ejection accident methodology for Fort Calhoun has been previously approved.

Since the main steamline break event was not benchmarked using CENTS, CENTS is not approved for licensing application for the analysis of a main steamline break event at this time.

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. Letter from M. J. Virgilio (NRC) to S. A. Toelle (ABB-CE), Acceptance for Referencing of Licensing Topical Report CE-NPD 282-P, "Technical Manual for the CENTS Code" (TAC No. M82718), March 17, 1994.

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

4. Letter from D. L. Wigginton (NRC) to W. G. Gates (OPPD), Loss of Coolant Accident Analyses for Fort Calhoun Station, Unit 1, (TAC No. M81831), March 26, 1992.

5. Letter from W. C. Walker (NRC) to W. G. Gates (OPPD), Steam Generator Tube Rupture Methodology, (TAC No. 66801), May 22, 1991.

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