UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 160 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated September 1, 1995, as supplemented by letters dated October 17 and November 15, 1995, Duke Power Company (DPC or the licensee), submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would revise TS 6.9.1.9 to include references to updated or recently approved methodologies used to calculate cycle-specific limits contained in the Core Operating Limits Report (COLR). The subject references have previously been reviewed and approved by the NRC staff.

The October 17 and November 15, 1995, letters provided clarifying information that did not change the scope of the September 1, 1995, application and the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

Specifically, the proposed amendments would provide corrected report numbers and dates on which the reports were approved by the NRC for six items in the list of approved references in TS 6.9.1.9, as listed below:

4. BAW-10168P, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," SER dated January 1991 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

5. DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March 1990 (DPC Proprietary).

(Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5- Shutdown Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1- Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3/4.9.12 - Spent Fuel Pool Boron Concentration.)

8. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," November 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P-A, "Thermal-Hydraulic Transient Analysis Methodology," August, 1994.

   (Modeling used in the system thermal-hydraulic analyses)

   This report was approved for use on the Catawba and McGuire Nuclear Stations by letter from T. A. Reed, NRC, to H. B. Tucker, DPC, dated November 15, 1991, subject to conditions stated in the Safety Evaluation. The August 1994 version of this report was transmitted by letter from M. S. Tuckman, DPC, to the NRC dated August 8, 1995.

The corrected report numbers and dates for the six references listed above are consistent with those in previously published NRC staff safety evaluations for these reports, and are therefore acceptable.

The proposed amendments also would add the following five reports (Items 11 through 15) to the list of NRC-approved references in TS 6.9.1.9.

11. DPC-NE-2004P-A. "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

The staff issued its evaluation of the Duke Power Company Topical Report DPC-NE-2004P-A, "Core Thermal-Hydraulic Methodology using VIPRE-01" by letter from T. A. Reed, NRC, to H. B. Tucker, DPC, on November 15, 1991. This report documents the use of the VIPRE-01 computer code and the statistical core design (SCD) methodology for the McGuire and Catawba core thermal-hydraulic analyses. The methodology was used for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor FdeltaH(X,Y). The staff concluded that the report is acceptable for referencing in the core thermal-hydraulic analyses for the McGuire and Catawba Nuclear Stations subject to the conditions stated therein. On these bases, the staff finds this proposed revision to TS 6.9.1.9, to add report DPC-NE-2004P-A, as approved, to be acceptable.

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW fuel," October 1990 (DPC Proprietary).

The staff issued its evaluation of the Duke Power Company Topical Report DPC-NE-2001P-A, Revision 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel" by letter from K. N. Jabbour, NRC, to H. B. Tucker, DPC, on October 15, 1990. This report documents DPC's mechanical reload analysis methodology for Mark-BW fuel to ensure the fuel rod structural integrity and to establish acceptable thermal and mechanical operating limits for the Catawba and McGuire Nuclear Stations. The methodology was used for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints. The staff concluded that the report is acceptable for Mark-BW fuel licensing applications in McGuire and Catawba and is limited to the use of the TACO2 code. On these bases, the staff finds this proposed revision to TS 6.9.1.9, to add report DPC-NE-2001P-A, Revision 1, to be acceptable.

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

The staff issued its evaluation of the Duke Power Company Topical Report DPC-NE-2005P-A, "Thermal Hydraulic Statistical Core Design Methodolog[illegible] letter from G. M. Holahan, NRC, to H. B. Tucker, DPC, on February 27, 19[illegible] This report documents the development of core thermal-hydraulic analysis based upon the statistical core design methodology using the VIPRE-01 computer code for the Catawba, McGuire and Oconee Nuclear Stations. The methodology was for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specification 3.2.1 - Axial Flux Difference, and Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor. The staff concluded that the report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation. On these bases, the staff finds this proposed revision to TS 6.9.1.9, to add report DPC-2005P, as approved, to be acceptable.

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

The staff issued its evaluation of the Babcock & Wilcox (B&W) Topical Report BAW-10162P-A, "TACO3 - Fuel Pin Thermal Analysis Computer Code" by letter from A. C. Thadani, NRC, to J. H. Taylor, B&W, on August 14, 1989. The revised TACO3 code addressed by this evaluation was developed to provide predictions of the thermal and mechanical performance of pressurized water reactor fuel rods experiencing variable power histories up to a particular burnup level. The staff's review concluded that the Topical Report provided an acceptable basis for changes to the B&W TACO3 computer code. The methodology was used for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February, 1994.

The staff issued its evaluation of the Babcock & Wilcox Topical Report BAW-10183P, "Fuel Rod Gas Pressure Criterion" by letter from A. C. Thadani, NRC, to J. H. Taylor, B&W, on February 22, 1994. The BAW-10183P report describes a fuel rod gas pressure criterion that the B&W Fuel Company (BWFC) would apply to existing fuel designs to allow the rod pressure to exceed system pressure under certain conditions. The staff's review concluded that the Topical Report provides an acceptable basis for the fuel rod gas pressure criterion for licensing applications. The criterion was used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints.

By letter dated May 4, 1994, DPC requested that NRC review and approve the transfer of the fuel performance code TACO3 from the BWFC to DPC for reload licensing applications. The transfer includes the approved Topical Reports BAW-10162P-A and BAW-10183P, as approved. The NRC staff concluded, in a letter from H. N. Berkow, NRC, to M. S. Tuckman, DPC, dated April 3, 1995, that DPC has the technical capability to perform TACO3 analyses for reload licensing applications and therefore, the use of TACO3 by DPC for the Catawba, McGuire, and Oconee Nuclear Stations is acceptable.

The proposed revision in DPC's application of September 1, 1995, adds the references for the two methodology reports, BAW-10162P-A and BAW-10183P, as approved, to TS 6.9.1.9. On the basis of its review of the licensee's submittals summarized above, the NRC staff finds that this proposed revision to TS 6.9.1.9 is acceptable.

Technical Specification 6.9.1.9 lists the core operating limits which shall be established and documented in the COLR before each reload cycle. It also lists the topical reports which provide the methodologies approved by the NRC for use in calculating the core operating limits. In the descriptions of the topical reports given in this listing, it may be noted that, for a number of core operating limits, several different topical reports (methodologies) have been approved for calculating the same operating limit. To meet the intent of

TS 6.9.1.9, the COLR should specify, for each fuel cycle, which topical report and which methodology is used to calculate each core operating limit. With this condition, the staff finds that the proposed revisions to TS 6.9.1.9. are acceptable as the use of NRC-approved methodologies will ensure that values for cycle-specific parameters are established consistent with applicable design bases and safety limits.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The amendments also change recordkeeping, reporting, or administrative procedures or requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 54718 dated October 25, 1995). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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