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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NPF-35

AND AMEDMENT NO. 31 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY ET AL.

DOCKET NOS. 50-413 AND 50-414

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

INTRODUCTION

By letter dated November 13, 1987 (Ref. 1), Duke Power Company, et al., (the licensee) requested Changes to the Technical Specifications (TSs) for Catawba Nuclear Station Units 1 and 2, to reflect the Unit 2 refueling and the addition of the Boron Dilution Mitigation System for Unit 2. In addition changes to TSs 4.3.3.12.1(b); 3.9.2.1, Actions (a)(2) and (d); and 4.9.1.3 are requested for both units. A supplemental letter dated December 11, 1987 (Ref. 2) provided a discussion of the Justification and No Significant Hazards Considerations. Additional information and justification were provided in letters dated January 15 and 20, 1988 (Refs. 9 and 10).

The substance of the changes noticed in the Federal Register on December 30, 1987 and the proposed no significant hazards determination were not affected by the licensee's letters dated January 15 and 20, 1988 which clarified certain aspects of the request.

EVALUATION

A. Unit 2 Cycle 2 Reload

1. General Design

The Catawba Unit 2, Cycle 2, reactor core contains 193 Optimized Fuel Assemblies. During the Cycle 1/2 refueling 64 Region 1 fuel assemblies will be replaced with 64 Region 4 fuel assemblies. The Region 4 fuel is very similar to that used in Regions 1, 2, and 3. Region 4 fuel assemblies have a smaller rod plenum spring than those used in Regions 1, 2, and 3. This new spring design is being generally incorporated by Westinghouse and the justification was submitted in Reference 3. The Region 4 fuel has been designed according to the fuel performance model in WCAP 8785 (Ref. 4). The fuel is designed and operated so that clad flattening will not occur as provided by the Westinghouse model in WCAP 8377 (Ref. 5). For all fuel regions, the fuel rod internal pressure design basis, which is discussed and shown acceptable in WCAP-8964 (Ref. 6) is satisfied.

8803030280 880216
PDR ADOCK 05000413
P PDR

The licensee provided a Reload Safety Evaluation (RSE) for Catawba 2, Cycle 2, as an attachment to Reference 1. The RSE presents a Cycle-specific evaluation for Cycle 2 which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was performed utilizing the approved reload design methods of WCAP-9273-U-A (Ref. 7).

2. Nuclear Design

The Cycle 2 Core loading is designed to meet an $[F_Q(Z)xP]$ ECCS limit of less than or equal to 2.32xK(Z). Adherence to the F_Q limit is obtained by using the F_Q TS surveillance described in WCAP-10217-A (Ref. 8). F_Q surveillance is part of the Relaxed Axial Offset Control (RAOC) and replaces the previous F_{xy} surveillance by comparing a measured F_Q limit. This provides a more convenient form of assuring plant operation below the F_Q limit while retaining the intent of using a measured parameter to verify operation below TS limits. The above discussion is consistent with Reference 8 which was approved. Thus, the staff finds that the TS change to F_Q surveillance is acceptable.

RAOC will be employed in Cycle 2 to enhance operational flexibility during non-steady state operation. RAOC makes use of available margin by expanding the allowable delta I band, particularly at reduced power. ROAC is described in Reference 8 and was approved by the staff. Thus, it is acceptable for use in Catawba Unit 2.

During operation at or near steady state equilibrium conditions, core peaking factors are significantly reduced due to the limited amount of xenon skewing possible under these operating conditions. The licensee proposes to use Base Load TSs to recognize this reduction in core peaking factors. The proposed Base Load TSs are identical to those that the staff has previously approved for McGuire Units 1 and 2, and Catawba Unit 1 and are therefore acceptable.

The RSE provides a table of Cycle 2 kinetics characteristics which are compared with the current limits based on previously approved accident analyses. The RSE also provides a table showing the results of the calculated Cycle 2 control rod worths and requirements at the most limiting condition during the cycle (end-of-life). These results include a standard 10% allowance for calculational uncertainty. From this information, the staff concludes that sufficient control rod worth will be available to provide the required shutdown margin for Cycle 2 operation. Control rod insertion limits were increased for less than 100% power for Cycle 2. Since the required shutdown margin is maintained, the TS change proposed to reflect the increased insertion is acceptable.

3. Thermal and Hydraulic Design

The thermal hydraulic methodology, DNBR correlation and core DNB limits used for Cycle 2 are consistent with the current licensing basis described in the FSAR and approved by the staff.

The power distributions produced by the cycle-specific RAOC analysis were analyzed for normal operation and Condition II events. Limits on the allowable operating flux difference as a function of power level from these considerations were found to be less restrictive than those resulting from LOCA F_Q considerations. The Condition II analyses generate DNB core limits and resultant overtemperature delta T setpoints. These generated a change to the $F(\Delta I)$ function in the TSs. The change is acceptable because it results from cycle-specific calculations using approved methods (Refs. 7 and 8). Therefore, the staff concludes that the Cycle 2 thermal-hydraulic analysis is acceptable.

4. Accident Analysis

The effects of the reload on the design basis and postulated accidents analyzed in the FSAR were examined. In all cases it was found that the effects were accommodated within the conservatism of the initial assumption used in the previous applicable safety analysis as well as those performed in support of the RTD Bypass removal and the UHI deletion (Refs. 1, 2 and 9). A core reload can affect accident analysis input parameters through control rod worths, core peaking factors and core kinetic characteristics. The Cycle 2 parameters in each of these areas were examined and found to be within the bounds of the current limits. Therefore, the staff concludes that the accident analysis is acceptable.

5. Technical Specification Changes

The Technical Specification changes for the Unit 2 Cycle 2 Reload are:

1. RAOC and Axial Flux Difference Limits
2. F_Q Surveillance
3. Base Load TSs
4. Rod Insertion Limits
5. $OT\Delta T\ f_1\ (\Delta I)$

Acceptability of items 1 - 4 was discussed in Section 2, Nuclear Design. Acceptability of item 5 was discussed in Section 3, thermal and hydraulic design. The proposed changes are for Unit 2 only but Unit 1 is included only administratively because the TSs for both units are combined in one document. The revisions to the bases are also acceptable.

B. Boron Dilution Mitigation System

1. Introduction

The Boron Dilution Mitigation System (BDMS) which is being installed in Unit 2 is the same as the BDMS which was installed in Unit 1. The BDMS was described in letters dated June 6, 1986 and September 9, 1986 (Refs. 11 and 12).

2. Technical Specification Changes

The changes for Unit 2 which deal with the (BDMS) are to TSs 4.1.1.1.3; 4.1.1.1.4; 4.1.1.2.2; Table 3.3-1, item 6.b; Table 3. 3-1, Action 5; Table 4.3-1, Note (9); 3/4.3.3.12; and 3/4.9.2. Changes to TSs 4.3.3.12.1(b); 3.9.2.1, Actions (a)(2) and (d); and 4.9.1.3 apply to both Units. Each change is discussed below.

The changes that apply to Unit 2 only are identical to those approved for Unit 1 TSs when the (BDMS) was installed in that Unit. The licensee requested that these changes not apply to Unit 2 until after the BDMS system has been calibrated, tested and declared operable. Furthermore, licensee stated (Ref. 10) that all the TSs applicable to boron dilution accidents which are to be deleted, will be adminstratively maintained in this interim period. The staff finds this acceptable.

TS 4.3.3.12.1(b)

This TS will be deleted because it is required only prior to Mode 2 but the specification itself is not applicable in Modes 1 and 2. The staff finds this change acceptable.

TS 3.9.2.1, Actions (a)(2) and (d)

This change to Action (a)(2)is an editorial change which deletes a phrase "and control room" which appeared twice in the sentence. Thus it is acceptable. The addition of Action (d) would allow the plant to change modes if the BDMS is inoperable. This statement already appears in TS 3.3.3.12 which covers all other applicable modes.

TS 4.9.1.3

This TS verifies that potential boron dilution flow paths are isolated when the unit is in Mode 6. The deletion of TS 4.9.1.3 is acceptable because the BDMS provides for automatic isolation of potential boron dilution flow paths.

ENVIRONMENTAL CONSIDERATION

These amendments involve changes to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve no significant increase in the ammounts, 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 exposures. The NRC staff has made a determination that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 49225) on December 30, 1987. The Commission consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

(1) Letter form Hal B. Tucker (Duke Power Company) to NRC, November 13, 1987

(2) Letter from Hal B. Tucker (Duke Power Company) to NRC, December 11, 1987

(3) Letter from E. P. Rahe Jr. (Westinghouse) to L. E. Phillps (NRC) April 12, 1984, NS-EPR-2893, Subject: Fuel Handling Load Curtain (6g vs 4g)

(4) Muller, J. V. (Ed.) "Improved Analytical Model used in Westinghouse Fuel Rod Design Computations, "WCAP-8785, October 1976

(5) George, R. A., (et al.), "Revised Clad Flattening Model," WCAP-8377, July 1977

(6) Risher, D. H., (et al.), "Safety Analyses for the Reused Fuel Rod Internal Pressure Design Basis," WCAP-8964, June 1977

(7) Bordelon, F. M., (et al.), "Westinghouse Reload Safety Evaluation Methodology", WCAP-9273 U-A, July 1985

(8) Muller, R. W. , (et al.), "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification, "WCAP-10217-A, June 1983

(9) Letter from Hal B. Tucker (Duke Power Company) to NRC, January 15, 1988

(10) Letter from Hal B. Tucker (Duke Power Company) to NRC, January 20, 1988

(11) Letter from Hal B. Tucker (Duke Power Company) to Harold Denton (NRC), June 6, 1986

(12) Letter from Hal B. Tucker (Duke Power Company) to Harold Denton (NRC), September 9, 1986

Principal Contributors: K. Jabbour, PDII-3/DRPI/II
M. Chatterton, DEST/SRXB

Dated: February 16, 1988