June 13, 1994

Docket Nos. 50-413
and 50-414

Mr. David L. Rehn
Vice President, Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

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R.Martin	OC/LFDCB T-9 E10
L.Berry	J.Johnson, RII
OGC 15B18	T. Attard, 8E23

Dear Mr. Rehn:

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 2 AND 2, (TAC NOS. M89127 AND M89126)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 120 to Facility Operating License NPF-35 and Amendment No. 114 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated March 24, 1994, as supplemented April 11 and May 31, 1994.

The amendments revise the TS to increase boron concentration for the spent fuel storage pool during Modes 1-3 operation and for the refueling canal during Mode 6 operation; include two reload related topical reports in TS 6.9.1.9; and correct errors in nomenclature and remove obsolete footnotes.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

Robert E. Martin, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 120 to NPF-35
2. Amendment No. 114 to NPF-52
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDII-3/LA	PDII-3/PM	OGC		PDII-3/D
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DATE	/ /94	6/7/94	6/9/94		6/10/94

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

June 13, 1994

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Mr. David L. Rehn
Vice President, Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

Dear Mr. Rehn:

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 1 AND 2, (TAC NOS. M89127 AND M89126)

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Robert Martin

Robert E. Martin, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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3. Safety Evaluation

cc w/enclosures:
See next page

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Catawba Nuclear Station

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

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. NPF-35

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company, acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc. (licensees), dated March 24, 1994, as supplemented April 11 and May 31, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-35 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 120, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert E Martin
for David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: June 13, 1994

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

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 114
License No. NPF-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated March 24, 1994, as supplemented April 11 and May 31, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-52 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 114 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for Robert E Martin

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: June 13, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 114

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
3/4 5-2	3/4 5-2
3/4 5-3	3/4 5-3
3/4 7-39	3/4 7-39
3/4 7-41	3/4 7-41
3/4 A9-1	-
3/4 B9-1	3/4 9-1
B 3/4 9-1	B 3/4 9-1
6-19a	6-19a

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2) The volume weighted average boron concentration of the accumulators less than the lower LCO limit presented in the Core Operating Limits Report but greater than the minimum required to ensure post-LOCA subcriticality presented in the Core Operating Limits Report, restore the inoperable accumulator to OPERABLE status or return the volume weighted average boron concentration of the accumulators to greater than the lower LCO limit presented in the Core Operating Limits Report and enter ACTION c.1 within 6 hours of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

3) The volume weighted average boron concentration of the accumulators equal to the minimum required to ensure post-LOCA subcriticality presented in the Core Operating Limits Report or less, return the volume weighted average boron concentration of the accumulators to greater than the minimum required to ensure post-LOCA subcriticality presented in the Core Operating Limits Report and enter ACTION c.2 within 1 hour of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

   1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

   2) Verifying that each cold leg injection accumulator isolation valve is open.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 75 gallons by verifying the boron concentration of the accumulator solution;

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 31 days when the Reactor Coolant System pressure is above 2000 psig by verifying that power is removed from the isolation valve operators on Valves NI54A, NI65B, NI76A, and NI88B and that the respective circuit breakers are padlocked; and

d. At least once per 18 months by verifying that each cold leg injection accumulator isolation valve opens automatically under each of the following conditions:

1) When an actual or a simulated Reactor Coolant System pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

2) Upon receipt of a Safety Injection test signal.

4.5.1.2 Each cold leg injection accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS

4.7.12 The groundwater level shall be determined at the following frequencies by monitoring the water level and by verifying the absence of alarm in the six groundwater monitor wells as shown in FSAR Figure 2-60 installed around the perimeter of the Reactor and Auxiliary Buildings:

a. At least once per 7 days when the groundwater level is at or below the top of the adjacent floor slab, and

b. At least once per 24 hours when the groundwater level is above the top of the adjacent floor slab.

b. At least once per 92 days by verifying that the individual cell voltage is greater than or equal to 1.36 volts on float charge, and

c. At least once per 18 months by verifying that:

1) The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

2) The battery-to-battery and terminal connections are clean, tight, and free of corrosion.

4.7.13.3 The Standby Makeup Pump water supply shall be demonstrated OPERABLE by:

a. Verifying at least once per 7 days:

1) That the requirements of Specification 3.9.10 are met and the boron concentration in the storage pool is greater than or equal to 2175 ppm, or

2) That a contained borated water volume of at least 112,320 gallons with minimum boron concentration of 2175 ppm is available and capable of being aligned to the Standby Makeup Pump.

b. Verifying at least once per 92 days that the Standby Makeup Pump develops a flow of greater than or equal to 26 gpm at a pressure greater than or equal to 2488 psig.

4.7.13.4 The Standby Shutdown System 250/125-Volt Battery Bank and its associated charger shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) That the electrolyte level of each battery is above the plates, and

2) The total battery terminal voltage is greater than or equal to 258/129 volts on float charge.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery, and

c. At least once per 18 months by verifying that:

1) The batteries, cell plates, and battery racks show no visual indications of physical damage or abnormal deterioration, and

2) The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met either:

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2175 ppm.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2175 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2175 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Boron Dilution Mitigation System ensures that monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY of the Reactor Building Containment Purge System ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and activated carbon adsorbers prior to release to the atmosphere. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Operation of the Reactor Building Containment Purge System and the resulting iodine removal capacity are consistent with the assumption of the safety analysis. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

CORE OPERATING LIMITS REPORT (Continued)

7. DPC-NF-2010P-A, "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.)

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, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," November 1991.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Design Methodology Using CASMO-3/Simulate-3P," November 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

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

    (Methodology 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 $F\Delta H$ (X,Y).)

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

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC in accordance with 10 CFR 50.4.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated March 24, 1994, as supplemented April 11 and May 31, 1994, Duke Power Company, et al. (the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would revise the TS to increase boron concentration for the spent fuel storage pool during Modes 1-3 operation and for the refueling canal during Mode 6 operation; include two reload related topical reports in TS 6.9.1.9; and correct errors in nomenclature and remove obsolete footnotes. The April 11, 1994, letter proposed corresponding changes to the BASES section. The May 31, 1994, submittal added two additional DPC topicals used in reload analysis methodology and references TS parameters determined using this methodology.

The April 11 and May 31, 1994, letters provided clarifying and additional information that did not change the scope of the March 24, 1994, application and the initial proposed no significant hazards consideration determination.

The reload report submitted on March 24, 1994 (Ref.1) for Catawba Unit 2, Cycle 7 (C2C7) contains TS changes, changes to the core operating limits report (COLR), markups of the appropriate FSAR chapters, and design information relative to the cycle 7 reload. Catawba Unit 2 recently operated in cycle 6 with a complete batch of B&W Mark-BW 17x17 fuel design. C2C7 will include a second complete batch of B&W Mark-BW 17x17 fuel.

The core consists of 193 assemblies containing 264 fuel rods, 24 guide tubes, and 1 incore instrument tube. The C2C7 core has 105 burned assemblies and 88 fresh assemblies. Data provided by the licensee shows that the 88 fresh assemblies will be loaded into the core in a symmetric checkerboard pattern. Similar changes reflecting the use of Mark-BW fuel and Duke methodology have been submitted and approved for the operation of Catawba Nuclear Station as Amendment 112 for Unit 1.

The reload design and all the analysis for normal and off-normal operations will be carried out inhouse by DPC. The methods and analytical models used by DPC for C2C7 fuel assembly mechanical design, nuclear design, thermal-hydraulic analyses, and non-LOCA safety analysis have been approved by the NRC (Refs. 2 to 6).

2.0 EVALUATION

2.1 Mechanical Design

The Cycle 7 reload will be the second time that the Mark-BW fuel will be used in Unit 2. This fuel is similar to the Westinghouse standard assembly design. The core consists of 88 fresh Mark-BW fuel assemblies. A total of forty eight (48) of these fresh assemblies will be natural uranium blanketed. Forty (40) of the blanketed assemblies will have an enrichment of 4.0 wt % U_{235} in the non-blanket region while the remaining eight blanketed assemblies will employ an enrichment of 3.60 wt % U_{235}. Forty fresh fuel assemblies will be non-blanketed, and the re-inserted fuel assemblies in Cycle 7 will be Westinghouse Optimized fuel assemblies and 29 Mark-BW fuel assemblies.

The mechanical analyses and thermal performance for the Mark-BW 17x17 design were performed by DPC with the methodology described in the approved topical report DPC-NE-2001-P-A, Revision 1 (Ref. 7); and therefore, are acceptable.

3.0 FUEL SYSTEM DESIGN

3.1 Fuel Management

A general description of the C2C7 core is given in section 3.0. The C2C7 core uses a low-leakage fuel management scheme where previously burned assemblies are placed on the periphery and most of the fresh assemblies are located throughout the core interior in a pattern which minimizes power peaking. With this loading and a cycle 6 endpoint of 380 EFPD, the cycle 7 reactive lifetime for full power operation is expected to be 430 EFPD. A comparison of cycle 7 nominal characteristic physics parameters with those used in the safety analyses show that the latter are conservative in all cases.

3.2 Nuclear Design

The core physics parameters for cycle 7 were generated similarly to those for cycle 6, using computer codes CASMO-3/SIMULATE-3P and the methodology as described in the approved topical reports DPC-NE-1004A and DPC-NE-3001-PA (Refs. 8 and 4). The Reactor Protection System setpoint limits and technical specification operating limits for the core were verified through analysis of the cycle 7 nuclear design using methodology described in the approved topical report DPC-NE-2011-P-A (Ref.9). The SIMULATE-3P calculations were performed in three dimensions.

These topical reports describe the physics analysis for determining safety related parameters pertaining to power distribution, reactivity worth and coefficients, and reactor kinetics characteristics. The DPC-NE-3001-PA report describes the methodology used by the licensee to ensure that the accident analysis for a defined reference core conservatively bounds the reload core. The Catawba 2 cycle 7 reload core physics parameters were reviewed with respect to the assumptions used in these analyses. The analysis and methodology for these events have been reviewed and found acceptable by the staff.

3.3 Control Requirement

The value of the required shutdown margin varies throughout core life with the most restrictive value occurring at end of cycle (EOC) and at hot zero power (HZP) conditions. Sufficient boration capability and net available control element assembly (CEA) worth, including a minimum worth stuck CEA and appropriate calculation uncertainties, exist to meet all the shutdown margin requirements. These results were derived by approved methods and incorporate appropriate assumptions and are, therefore, acceptable, (Ref. 9).

4.0 THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic analyses supporting cycle 7 operation was performed by DPC with VIPRE-01 computer code and approved statistical core design (SCD) methodology (Refs. 2 and 10). The statistical core design methodology is a technique that statistically combines uncertainties associated with the core statepoint parameters, code/model, and Critical Heat Flux (CHF) correlation to determine a statistical DNBR limit (SDL). The uncertainties used in Reference 2 bound the uncertainties specifically calculated for Catawba Unit 2. The statistical DNBR limit for use with the BWCMV CHF correlation (Ref. 10) in VIPRE-01 is determined to be 1.40. To provide design flexibility, a 10.7% margin is added to the statistical DNBR limit to yield a design DNBR Limit of 1.55 for the generic Mark-BW and the Catawba Unit 2 cycle 7 analyses. The reactor core safety limits for Catawba Unit 2 cycle 7 were generated utilizing the BWCMV CHF correlation and the SCD methodology for a full core of Mark-BW assemblies and a radial enthalpy rise hot channel factor of 1.50.

The hydraulic compatibility of the Mark-BW fuel and the Westinghouse Optimized Fuel Assemblies (OFA) had been addressed in the approved topical report BAW-10173-P-A Revision 2 (Ref. 11). The results of the hydraulic compatibility test indicated that the total pressure drop across the Mark-BW Fuel is 2.4% lower than the total pressure drop across the OFA fuel. The licensee approach to addressing the transition core penalty is presented in detail in Reference 5. The licensee determined a generic transition core penalty by modelling a conservative core configuration with one OFA assembly as the hot assembly located in a Mark-BW core. Bounding power shapes during normal and accident conditions were analyzed yielding a maximum DNBR penalty of 3.8 % for OFA fuel. The Licensee addressed the transition core penalty for OFA fuel by applying the 3.8% DNBR penalty against the 10.7% generic margin included in the design DNBR limit.

Prior to Catawba 2 cycle 7, a 2.8% DNBR penalty was applied against the margin in the design DNBR limit to account for the flow distribution effects of the grid restrain system used for MK-BW fuel assemblies. The licensee stated that this penalty was conservatively estimated using VIPRE-01. The licensee also stated that the B&W Fuel Company (BWFC) has performed several CHF tests which show that this 2.8% DNBR penalty is not required for the system used to hold the intermediate grids in place. BWFC has submitted the results of the CHF test in a topical report titled "BAW-10189 Mixing Vane CHF correlations". The NRC staff is currently reviewing this topical, and until such time as the staff has completed the review, the 2.8% DNBR penalty will continue to apply to Catawba.

5.0 ACCIDENT ANALYSES

5.1 NON-LOCA SAFETY ANALYSIS

The design basis events (DBEs) considered in the safety analyses are categorized into two groups: anticipated operational occurrences (AOOs) and postulated accidents (limiting faults). All events were reviewed by the licensee to account for the differences in the core physics parameters of the Mark-BW fuel and the changes to the Technical Specifications. The scope of the events considered is consistent with that addressed in the existing Final Safety Analysis Report (FSAR) for Catawba. The evaluations considered the effects of mixed (transition) cores using Westinghouse and Mark-BW fuel.

The Licensee analyzed the rod drop event using cycle specific axial flux shape. The result of the analysis indicated that the existing limiting case was unchanged with the slight increase in peaking of the axial flux for C2C7, and that the case remains limiting.

The axial blanketed fuel used in the C2C7 reload requires the allocation of 3.0% DNBR margin for DNB analyses. This DNBR penalty will account for the potential non-conservative behavior of the axial power distribution in the blanketed fuel assemblies. Data provided by the licensee showed DNBR penalties assessed against available margin.

The post-LOCA subcriticality analysis required an increase in the refueling water storage tank (RWST) minimum boron concentration from 2000 ppm to 2175 ppm and an increase in the cold leg accumulator (CLA) from 1900 ppm to 2000 ppm. The maximum RWST and CLA limits are accordingly increased to preserve an operating margin. These changes are reflected in TS 4.7.13.3 and 3.9.1.

In addition, the increase in RWST and CLA maximum boron concentration limits necessitated a reanalysis of the post-LOCA boron precipitation evaluation and of the post-LOCA containment sump pH. The post-LOCA boron precipitation analysis required a reduction in the time that the operator must initiate recirculation through the hot leg from 9 hours to 7 hours. The post-LOCA sump pH analysis indicated that the existing range in the TS BASES is acceptable.

The licensee has also found that the slope and breakpoint of the over-temperature delta-temperature reactor trip function has been reevaluated for cycle 7 reload design. The reevaluation resulted in the removal of some of the conservatism in the C2C7 reload design. These analyses were performed using methodologies as described in the topical reports listed as references in the licensee's application. These topical reports have been reviewed previously by the NRC staff and have been found acceptable as stated in the safety evaluations reports. The staff's review of C2C7 reload parameters found them to be bounded by the accident analysis assumptions stated by the licensee, and are therefore acceptable.

5.2 LOCA ANALYSES

The LOCA analyses for Catawba Unit 2 transition cores with mixed Mark-BW and Westinghouse OFA assemblies and future cores with all Mark-BW fuel have been reviewed previously by the NRC and found acceptable.

6.0 TECHNICAL SPECIFICATION (TS) CHANGES

(1) Revision to TS 4.7.13.3

The change to TS 4.7.13.3 will increase the spent fuel storage pool minimum boron concentration from 2000 ppm to 2175 ppm. This is a conservative revision to the TS and is required to maintain consistency between the boron concentration of the spent fuel pool and the boron concentration of the RWST during modes 1 through 3 operation. Since TS 4.7.13.3 is used in regard to TS 3.1.2.6, this change will make TS 4.7.13.3 consistent with TS 3.1.2.6. The revision to TS change 4.7.13.3 is acceptable.

(2) Changes to TS 3.9.1

The change to TS 3.9.1 will increase the reactor coolant system (RCS) and the refueling water canal minimum boron concentration from 2000 ppm to 2175 ppm. This revision is required to maintain consistency between the boron concentrations of the RCS and the refueling canal and the boron concentration of the RWST during mode 6 operation. The licensee pointed out that the unit specification designation for TS 3.9.1 has been removed as the concentrations at Units 1 and 2 are now identical. This change has already been approved for Unit 1 under amendment 112 issued December 17, 1993 for Catawba Unit 1 cycle 8. This change to TS 3.9.1 is acceptable.

(3) Changes to TS 6.9.1.9

The changes to TS 6.9.1.9 will reflect the inclusion of two reload related topicals, which have been previously reviewed and approved by the NRC. The two topicals, DPC-NE-2004P-A and DPC-NE-2001P-A, were added to the list of topical reports that are used to determine the core operating limits. (Ref.12).

The licensee also made administrative revisions to surveillance requirements SR 4.5.1.1 and SR 4.7.12.

REVIEW CONCLUSION

The NRC staff has reviewed the reports submitted by the licensee for the operation of Catawba Unit 2 cycle 7 and the material submitted in regard to TS and COLR changes pertaining to this reload. Based on this review, we have concluded that the requested TS changes satisfy staff positions and requirements in these areas.

The licensee's proposal to remove the 2.8% penalty associated with grid flow distribution effects from Table 6-3 of the licensee's reload report is not approved at this time since this is the subject of the staff's ongoing review of the topical report BAW-10189 Mixing Vane CHF Correlations.

7.0 STATE CONSULTATION

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

8.0 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 and change surveillance 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 (59 FR 22006 dated April 28, 1994). 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 the amendments.

9.0 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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REFERENCES

1. Duke Power Company Letter, D. L. Rehn, dated March 24, 1994, to U.S. Nuclear Regulatory Commission, Document control desk, Washington DC 20555.

2. DPC-NE-2004P-A, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRER-01, Duke Power Company, December 1991.

3. DPC-NE-3000P,Duke Power Company, Thermal-Hydraulic Transient Analysis Methodology, Revision 2, February 20, 1991.

4. DPC-NE-3001-PA, Duke Power Company, Multidimensional Reactor Transient Safety Analysis Parameters Methodology, Revision 2, November 1991.

5. BAW-10174P-A, MARK-BW Reload LOCA Analysis for Catawba and McGuire, Babcock & Wilcox, Revision 1, November, 1990.

6. DPC-NE-3002-A, McGuire Nuclear Station/Catawba Nuclear Station FSAR Chapter 15 System Transient Analysis Methodology, November 1991.

7. BAW-1072-PA, Mark-BW Mechanical Design report, Babcock and Wilcox, Lyncburg, Virginia, December 19, 1989.

8. DPC-NE-1004A, Nuclear Design methodology Using CASMO-3/SIMULATE-3P, Duke Power Company, November 1992.

9. DPC-NE-2011-PA, Nuclear Design Methodology for core operating limits of Westinghouse Reactors, Duke Power Company, March 1990.

10. BAW-10159P-A, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, Babcock & Wilcox, July 1990.

11. BAW-10173P-A, MARK-BW Reload Safety Analysis for Catawba and McGuire, Babcock & Wilcox, Revision 2, February 20, 1991.

12. Duke Power Company Letter, D. L. Rehn, dated May 31, 1994, to U.S. Nuclear Regulatory Commission, Document Control Desk, Washington DC 20555.