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

October 2, 2000

Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA8719 AND MA8720)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 188 to Facility Operating License NPF-35 and Amendment No. 181 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 June 29, 2000, as supplemented by letters dated July 27, and August 10, 2000. Other related information was submitted by letters dated April 10, April 17, and June 19, 2000.

The amendments revise TS to reference the Westinghouse Best Estimate Large Break Loss-of-Coolant Accident (LOCA) analysis methodology described in WCAP-12945-P-A, March 1998. These amendments also address corresponding TS Bases changes.

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,

Chandu P. Patel

Chandu P. Patel, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 188 to NPF-35
2. Amendment No. 181 to NPF-52
3. Safety Evaluation

cc w/encls: See next page

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/RA/

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Catawba Nuclear Station

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Catawba Nuclear Station

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

DUKE ENERGY CORPORATION

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. 188
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 Energy Corporation, acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc. (licensees), dated June 29, 2000, as supplemented by letters dated July 27, and August 10, 2000, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 188 , which are attached hereto, are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: October 2, 2000

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

DUKE ENERGY CORPORATION

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. 181
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 Energy Corporation, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated June 29, 2000, as supplemented by letters dated July 27, and August 10, 2000, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 181 , which are attached hereto, are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: October 2, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND LICENSE AMENDMENT NO. 181

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Appendix A Technical Specifications and associated Bases with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
5.6-5	5.6-5
B 3.2.1-2	B 3.2.1-2
B 3.2.1-5	B 3.2.1-5
B 3.2.1-11	B 3.2.1-11
B 3.2.2-2	B 3.2.2-2
B 3.2.4-1	B 3.2.4-1
B 3.5.1-3	B 3.5.1-3
B 3.5.1-4	B 3.5.1-4
B 3.5.1-5	B 3.5.1-5
B 3.5.2-3	B 3.5.2-3

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

13. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary).

14. DPC-NE-2009P-A, "Westinghouse Fuel Transition Report," SER dated September 22, 1999 (DPC Proprietary).

15. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2-5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998, (W Proprietary).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Ventilation Systems Heater Report

When a report is required by LCO 3.6.10, "Annulus Ventilation System (AVS)," LCO 3.7.10, "Control Room Area Ventilation System (CRAVS)," LCO 3.7.12, Auxiliary Building Filtered Ventilation Exhaust System (ABFVES)," LCO 3.7.13, "Fuel Handling Ventilation Exhaust System (FHVES)," or LCO 3.9.3, "Containment Penetrations," a report shall be submitted within the following 30 days. The report shall outline the reason for the inoperability and the planned actions to return the systems to OPERABLE status.

5.6.7 PAM Report

When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.8 Steam Generator Tube Inspection Report

a. The number of tubes plugged in each steam generator shall be reported to the NRC within 15 days following completion of the program;

(continued)

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F for small breaks and there is a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1);

b. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(X,Y,Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other Reference 1 criteria must also be met in LOCAs (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, transient strain, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(X,Y,Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(X,Y,Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(X,Y,Z)$ satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

The Heat Flux Hot Channel Factor, $F_Q(X,Y,Z)$, shall be limited by the following relationships:

$$F_Q^M(X,Y,Z) \le \frac{F_Q^{RTP}}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q^M(X,Y,Z) \le \frac{F_Q^{RTP}}{0.5} K(Z) \qquad \text{for } P \le 0.5$$

BASES

LCO (continued)

The $F_Q(X,Y,Z)$ limits typically define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a small break LOCA and a high level of probability that the peak cladding temperature does not exceed 2200°F for a large break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the $F_Q(X,Y,Z)$ limits. If $F_Q(X,Y,Z)$ cannot be maintained within the steady state LOCA limits, reduction of the core power is required.

Violating the steady state LOCA limits for $F_Q(X,Y,Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(X,Y,Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(X,Y,Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power. The exception to this is the steam line break event, which is assumed for analysis purposes to occur from very low power levels. At these low power levels, measurements of $F_Q(X,Y,Z)$ are not sufficiently reliable. Operation within analysis limits at these conditions is inferred from startup physics testing verification of design predictions of core parameters in general.

ACTIONS

<u>A.1</u>

Reducing THERMAL POWER by $\geq$ 1% RTP for each 1% by which $F^M_Q(X,Y,Z)$ exceeds its steady state limit, maintains an acceptable absolute power density. $F^M_Q(X,Y,Z)$ is the measured value of $F_Q(X,Y,Z)$ and the steady state limit includes factors accounting for measurement uncertainty and manufacturing tolerances. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

BASES

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M{}_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M{}_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M{}_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(X,Y,Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", March 1990.

BASES

BACKGROUND (continued)

uncontrolled RCCA bank withdrawal (UCBW). For these types of accidents, the event itself causes changes in the power distribution and this LCO alone is not sufficient to preclude DNB. The acceptability of analyses such as the UCBW accident analysis is ensured by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits," in combination with cycle-specific analytical calculations."

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs.

APPLICABLE SAFETY ANALYSES

Limits on $F_{\Delta H}(X,Y)$ preclude core power distributions that exceed the following fuel design limits:

a. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

b. During a large break loss of coolant accident (LOCA), there must be a high level of probability that the peak cladding temperature (PCT) does not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F_{\Delta H}(X,Y)$ are the core parameters of most importance. The limits on $F_{\Delta H}(X,Y)$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency that do not alter the core power distribution. For transients such as uncontrolled RCCA bank withdrawal, which are characterized by changes in the core power distribution, this LCO alone is not sufficient to preclude DNB. The acceptability of the accident analyses

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), there must be a high level of probability that the peak cladding temperature does not exceed 2200°F (Ref. 1);

b. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}(X,Y)$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators, safety injection pumps, and centrifugal charging pumps all play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 3) will be met following a small break LOCA and there is a high level of probability that the criteria are met following a large break LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute directly to the long term cooling requirements of 10 CFR 50.46. However, the boron content of the accumulator water helps to maintain the reactor core subcritical after reflood, thereby eliminating fission heat as an energy source for which cooling must be provided.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. The large and small break LOCA analyses are performed with accumulator volumes that are consistent with the LOCA evaluation models. To allow for operating margin, values of ± 30 ft^3 are specified.

The minimum boron concentration setpoint is used in the post LOCA sump boron concentration calculation. The calculation is performed to

BASES

APPLICABLE SAFETY ANALYSES (continued)

assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH. In particular, the equilibrium sump pH should be at least 7.5 following the design basis LOCA.

The large and small break LOCA analyses are performed with accumulator pressures that are consistent with the LOCA evaluation models. To allow for operating margin and accumulator design limits, a range from 585 psig to 678 psig is specified. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Ref. 4).

The accumulators satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met. Additionally, the nitrogen and liquid volumes between accumulators must be physically separate.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

BASES

APPLICABILITY (continued)

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F for small break LOCAs and there is a high level of probability that the peak cladding temperature does not exceed 2200°F for large break LOCAs.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break for the plant. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

BASES

BACKGROUND (continued)

The high and intermediate head subsystems of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a safety injection actuation.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a small break LOCA and there is a high level of probability that the criteria are met following a large break LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 181 TO FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated June 29, 2000, as supplemented by letters dated July 27, and August 10, 2000, Duke Energy Corporation, et al. (DEC, the licensee), submitted a request for changes to the Catawba Nuclear Station (Catawba), Units 1 and 2, Technical Specifications (TS). Other related information was submitted by letters dated April 10, April 17, and June 19, 2000. The requested changes would modify TS to reference the Westinghouse (W) Best Estimate Large Break Loss-of-Coolant Accident (LOCA) analysis methodology described in WCAP-12945-P-A, March 1998. The proposed changes also identify corresponding TS Bases changes. On June 12, 2000, DEC and W met with NRC staff to describe the method of applying the large break LOCA methodology to the four units of the McGuire and Catawba Nuclear Stations.

The letters dated April 10, April 17, June 19, July 27, and August 10, 2000, provided additional information that did not change the scope of the June 29, 2000, application and the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The NRC review covered the following:

a. Verifying that the W best estimate large break LOCA methodology applies to the four DEC plants,

b. Verifying the acceptability and application of the DPC proposed method of applying of the large break LOCA methodology to the four DEC units, and

c. Verifying that the proposed TS changes refer to the use of the W best estimate large break LOCA methodology.

2.1 LOCA Methodology Applies to McGuire/Catawba

The version of W best estimate large break LOCA methodology described in WCAP-12945-P-A was approved by NRC for application to W 3-loop and 4-loop plant designs. The McGuire, Units 1 and 2, and Catawba, Units 1 and 2 units, are of W four-loop design and have no design features that would invalidate use of the methodology. In the letter of August 10, 2000, the licensee described ongoing DEC and W processes which assure that analysis input values for parameters that are qualitatively or quantitatively significant to the results of analyses will bound the as-operated plant values for those parameters, or where appropriate, the ranges of such parameters input to the analyses will bound the as-operated plant values of those parameters.

We conclude that the W best estimate large break LOCA methodology described in WCAP-12945-P-A applies to the McGuire/Catawba plants, because: 1) the W best estimate large break LOCA methodology was approved for application to the McGuire/Catawba class of plants, 2) the licensee has identified ongoing processes which assure that input values to analyses using this methodology will be appropriate to represent the plants and are consistent with the methodology.

2.2 Adaptation of the W Best Estimate Large Break LOCA Methodology for Catawba and McGuire Licensing-Basis Analyses

The NRC safety evaluation report (SER) for the generic W best estimate large break LOCA methodology did not provide for multiple plant licensing-basis reference of a single bounding analysis using this methodology. DEC has proposed that the four DEC units reference a bounding licensing-basis analysis of a "composite" design incorporating the most adverse features among the four DEC units where the designs differ (slightly). However, in the meeting on June 12, 2000, the licensee provided information to show that, for the initial licensing-basis analysis, the "composite" plant analysis would be qualitatively representative and quantitatively bounding for all four DEC units.

To address concerns that for future changes the "composite" design analysis would not continue to represent and bound the McGuire and Catawba plants, the licensee and W described ongoing processes, which W and DEC would implement for each of the plants to identify and assess items that might change the representative and bounding nature of the "composite" plant analysis for any or all of the plants. Additionally, the licensee has committed to report on each plant separately and include a report of the "composite" plant analysis with each individual plant report. This will assure that the representative and bounding nature of the "composite" plant analysis could be confirmed or invalidating differences identified. Any such differing plant would be analyzed separately on a plant-specific basis. The licensee's letter dated August 10, 2000, provides a summary discussion of the ongoing processes to which DEC has committed.

The staff concludes that the programmatic provisions provided by the licensee give adequate assurance that the "composite" plant analysis will continue to be representative and bounding for the four DEC units, or that any significantly differing unit will be identified and analyzed on a plant specific basis.

2.3 Technical Specifications

In an attachment to the letter dated June 29, 2000, the licensee proposed changes to the Catawba TS and Bases to reflect the implementation of the W best estimate large break LOCA methodology as discussed in Sections 2.0 and 2.1. In the letter dated August 10, 2000, the licensee stated that the programmatic provisions discussed in Section 2.2 assure that the bounding "composite" analysis approach continues to apply to each of the units. The licensee committed to maintain and implement these provisions, as described in the letter dated August 10, 2000.

The staff concludes the W best estimate large break LOCA methodology described in WCAP-12945-P-A is acceptable for application to McGuire, Units 1 and 2 and Catawba, Units 1 and 2, because the licensee has demonstrated that the methodology applies to the McGuire/Catawba class of plants, as discussed in Section 2.0, and because the licensee has demonstrated that the adapted methodology applies specifically to the individual units through programmatic provisions, as discussed in Section 2.2. The staff also concludes that WCAP-12945-P-A is suitable for reference in McGuire and Catawba licensing documentation, including Technical Specifications and Core Operating Limit Reports.

4.0 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.

5.0 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 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 (65 FR 51350). 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.

6.0 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.

Principal Contributor: Frank Orr

Date: October 2, 2000