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. 115
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 January 13, 1993, as supplemented January 28 and April 26, 1993, 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. 115 , 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 and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

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: March 25, 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. 109
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 January 13, 1993, as supplemented January 28 and April 26, 1993, 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. 109 , 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 and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

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: March 25, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 115

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 109

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
2-A7 & 2-B7	2-7
2-A8 & 2-B8	2-8
2-A9 & 2-B9	2-9
2-A10 & 2-B10	2-10
3/4 A1-11 & 3/4 B1-11	3/4/1-11
3/4 A1-12 & 3/4 B1-12	3/4 1-12
3/4 3-85	3/4 3-85
3/4 3-86	3/4 3-86
3/4 A5-1 & 3/4 B5-1	3/4 5-1
3/4 A5-2 & 3/4 B5-2	3/4 5-2
3/4 5-3	3/4 5-3*
3/4 A5-11 & 3/4 B5-11	3/4 5-11
3/4 9-1a	3/4 9-1a
3/4 9-1b	3/4 9-1b
B 3/4 1-3	B 3/4 1-3
B 3/4 1-3a	B 3/4 1-3a
B 3/4 1-3b	B 3/4 1-3b
B 3/4 5-1	B 3/4 5-1
B 3/4 5-3	B 3/4 5-3
6-18	6-18
6-19	6-19
6-19a	6-19a*

* no change - overflow page

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right)\left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right)\left[T\left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3 (P - P') - f_1 (\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by Loop Narrow Range RTDs;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, as presented in the Core Operating Limits Report;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, as presented in the Core Operating Limits Report;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	Overtemperature ΔT reactor trip setpoint as presented in the Core Operating Limits Report;
K_2	=	Overtemperature ΔT reactor trip heatup setpoint penalty coefficient as presented in the Core Operating Limits Report;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, as presented in the Core Operating Limits Report;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, as presented in the Core Operating Limits Report;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

$T' \leq$ 590.8°F (Nominal T_{avg} allowed by Safety Analysis);

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient as presented in the Core Operating Limits Report;

P = Pressurizer pressure, psig;

P′ = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) For $q_t - q_b$ between the "positive" and "negative" $f_1(\Delta I)$ breakpoints as presented in the Core Operating Limits Report;

$f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent ΔI that the magnitude of $q_t - q_b$ is more negative than the $f_1(\Delta I)$ "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "negative" slope presented in the Core Operating Limits Report; and

(iii) For each percent ΔI that the magnitude of $q_t - q_b$ is more positive than the $f_1(\Delta I)$ "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "positive" slope presented in the Core Operating Limits Report.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.5% of Rated Thermal Power.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S)}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 = Overpower ΔT reactor trip setpoint as presented in the Core Operating Limits Report

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg}, as presented in the Core Operating Limits Report,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = Overpower ΔT reactor trip heatup setpoint penalty coefficient as presented in the Core Operating Limits Report for $T > 590.8°F$ and $K_6 = 0$ for $T \leq 590.8°F$,

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq 590.8°F$),

S = As defined in Note 1,

and f_2 (ΔI) is a function of the indicated differences between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between the "positive" and "negative" $f_2(\Delta I)$ breakpoints as presented in the Core Operating Limits Report; $f_2(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent ΔI that the magnitude of $q_t - q_b$ is more negative than the $f_2(\Delta I)$ "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent ΔI that magnitude of $q_t - q_b$ is more positive than the $f_2(\Delta I)$ "positive" breakpoint presented in the Core Operating Limits Report the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "positive" slope presented in the Core Operating Limits Report.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% (Unit 1) and 3.3% (Unit 2) of Rated Thermal Power.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume as presented in the Core Operating Limits Report,

2) A minimum boron concentration as presented in the Core Operating Limits Report, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1) A minimum contained borated water volume as presented in the Core Operating Limits Report,

2) A minimum boron concentration as presented in the Core Operating Limits Report, and

3) A minimum solution temperature of 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the refueling water storage tank temperature when it is the source of borated water and the outside air temperature is less than 70°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume as presented in the Core Operating Limits Report,

2) A minimum boron concentration as presented in the Core Operating Limits Report, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1) A minimum contained borated water volume as presented in the Core Operating Limits Report or Specification 3.5.4a whichever is larger,

2) A minimum boron concentration as presented in the Core Operating Limits Report,

3) A minimum solution temperature of 70°F, and

4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

BORON DILUTION MITIGATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.11 As a minimum, two trains of the Boron Dilution Mitigation System shall be OPERABLE and operating with Shutdown Margin Alarm ratios set at less than or equal to 4 times the steady-state count rate.

APPLICABILITY: MODES 3, 4, AND 5

ACTION:

(a) With one train of the Boron Dilution Mitigation System inoperable or not operating, restore the inoperable train to OPERABLE status within 48 hours, or

(1) suspend all operations involving positive reactivity changes and verify that valve NV-230 is closed and secured within the next hour, or

(2) verify two Source Range Neutron Flux Monitors are OPERABLE with Alarm Setpoints less than or equal to one-half decade (square root of 10) above the steady-state count rate and verify that the combined flowrate from both Reactor Makeup Water Pumps is less than or equal to the Reactor Makeup Water Pump Flowrate presented in the Core Operating Limits Report within the next hour.

(b) With both trains of the Boron Dilution Mitigation System inoperable or not operating, restore the inoperable trains to OPERABLE status within 12 hours, or

(1) suspend all operations involving positive reactivity changes and verify that valve NV-230 is closed and secured within the next hour, or

(2) verify two Source Range Neutron Flux Monitors are OPERABLE with Alarm Setpoints less than or equal to one-half decade (square root of 10) above the steady-state count rate and verify that the combined flow rate from both Reactor Makeup Water Pumps is less than or equal to the Reactor Makeup Water Pump Flowrate presented in the Core Operating Limits Report within the next hour.

SURVEILLANCE REQUIREMENTS

4.3.3.11.1 Each train of the Boron Dilution Mitigation System shall be demonstrated OPERABLE by performance of:

(a) A CHANNEL CHECK at least once per 12 hours,

SURVEILLANCE REQUIREMENTS (Continued)

(b) An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and

(c) At least once per 18 months the BDMS shall be demonstrated OPERABLE by:

(1) Verifying that each automatic valve actuated by the BDMS moves to its correct position upon receipt of a trip signal, and

(2) Verifying each reactor makeup water pump stops, as designed, upon receipt of a trip signal.

4.3.3.11.2 If using the Source Range Neutron Flux Monitors to meet the requirements of Technical Specification 3.3.3.11,

(a) The monthly surveillance requirements of Table 4.3-1 for the Source Range Neutron Flux Monitors shall include verification that the Alarm Setpoint is less than or equal to one-half decade (square root of 10) above the steady-state count rate.

(b) The combined flow rate from both Reactor Makeup Water Pumps shall be verified as less than or equal to the Reactor Makeup Water Pump Flowrate presented in the Core Operating Limits Report at least once per 31 days.

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open,

b. A contained borated water volume of between 7630 and 8079 gallons,

c. A boron concentration between the LCO limits presented in the Core Operating Limits Report,

d. A nitrogen cover-pressure of between 585 and 678 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve or boron concentration less than the lower LCO limit presented in the Core Operating Limits Report, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one accumulator inoperable due to boron concentration less than the lower LCO limit presented in the Core Operating Limits Report and:

 1) The volume weighted average boron concentration of the accumulators equal to the lower LCO limit presented in the Core Operating Limits Report or greater, restore the inoperable accumulator to OPERABLE status within 24 hours of the low boron determination or be in at least HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

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*Reactor Coolant System pressure above 1000 psig.

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 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.

4.5.2 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.

**This surveillance need not be performed until prior to entering HOT STANDBY following the Unit 1 refueling.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

a. A minimum contained borated water volume of 363,513 gallons,

b. A boron concentration of between the limits presented in the Core Operating Limits Report,

c. A minimum solution temperature of 70°F, and

d. A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The refueling water storage tank shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water level in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the refueling water storage tank temperature when the outside air temperature is less than 70°F or greater than 100°F.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2.1 As a minimum, two trains of the Boron Dilution Mitigation System shall be OPERABLE and operating with Shutdown Margin Alarm Ratios set at less than or equal to 4 times the steady-state count rate, each with continuous indication in the control room.

APPLICABILITY: MODE 6

ACTION:

(a) With one or both trains of the Boron Dilution Mitigation System inoperable or not operating,

(1) immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes, and verify that valve NV-230 is closed and secured within the next hour or

(2) verify that two Source Range Neutron Flux Monitors are OPERABLE and operating with Alarm Setpoints less than or equal to onehalf decade (square root of 10) above the steady-state count rate, each with continuous visual indication in the control room and one with audible indication in the control room and one with audible indication in the containment and verify that the combined flowrate from both Reactor Makeup Water Pumps is less than or equal to the Reactor Makeup Water Pump flowrate presented in the Core Operating Limits Report within the next hour.

(b) With both trains of the Boron Dilution Mitigation System inoperable or not operating and one of the Source Range Neutron Flux Monitors inoperable or not operating immediately suspend all operations involving core ALTERATIONS or positive reactivity changes and verify that valve NV-230 is closed and secured within the next hour.

(c) With both trains of the Boron Dilution Mitigation System inoperable or not operating and both of the Source Range Neutron Flux Monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours and verify that valve NV-230 is closed and secured within the next hour.

SURVEILLANCE REQUIREMENTS

4.9.2.1.1 Each train of the Boron Dilution Mitigation System shall be demonstrated OPERABLE by performance of:

(a) A CHANNEL CHECK at least once per 12 hours,

(b) An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS and

SURVEILLANCE REQUIREMENTS (Continued)

(c) An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days.

(d) At least once per 18 months the BDMS shall be demonstrated OPERABLE by:

(1) Verifying that each automatic valve actuated by the BDMS moves to its correct position upon receipt of a trip signal, and

(2) Verifying each reactor makeup water pump stops, as designed, upon receipt of a trip signal.

4.9.2.1.2 If using the Source Range Neutron Flux Monitors to meet the requirements of Technical Specification 3.9.2, each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

(a) A CHANNEL CHECK at least once per 12 hours,

(b) An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS or within 1 hour after declaring the BORON DILUTION MITIGATION SYSTEM inoperable, and

(c) An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

(d) The combined flowrate from both Reactor Makeup Water Pumps shall be verified as less than or equal to the Reactor Makeup Water Pump flowrate presented in the Core Operating Limits Report at least once per 7 days.

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions. To maintain shutdown margin for this condition a minimum water volume at a minimum boron concentration, as presented in the Core Operating Limits Report, is required from the boric acid storage tanks or the refueling water storage tank.

The Technical Specification requires a minimum contained water volume and boron concentration, as presented in the Core Operating Limits Report, be available from the boric acid tanks in Modes 1-4. This volume is based on the required volume for maintaining shutdown margin, unusable volume (to allow for a full suction pipe), instrument error, and additional margin for conservatism as follows:

Modes 1-4 Boric Acid Tank

Required volume for maintaining SDM	presented in the Core Operating Limits Report
Additional Margin	1,303 gallons
Unusable Volume (to maintain full suction pipe)	7,230 gallons
14" of water equivalent	
Vortexing (4" of water above top of suction pipe	2,066 gallons
Instrumentation Error (Based on Total Loop Acc. for 1&2 NV5740 loops) - 2" of water equivalent	1,550 gallons

A similar approach is taken for calculating the required Refueling Water Storage Tank volume:

When the temperature of one or more cold legs drops below 285°F in Mode 4, the potential for low temperature overpressurization of the reactor vessel makes it necessary to render one charging pump INOPERABLE and at least one safety injection pump INOPERABLE. The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 285°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

Refueling Water Storage Tank Requirements For Maintaining SDM - Modes 1-4

Required Volume for Maintaining SDM	presented in the Core Operating Limits Report
Unusable Volume (below nozzle)	13,442 gallons
Instrument Inaccuracy	11,307 gallons
Vortexing	13,247 gallons
Additional Margin	3,504 gallons

BORATION SYSTEMS (Continued)

With the coolant temperature below 200°F, one Boron Injection flow path is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection flow path becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. To maintain shutdown margin for this condition a minimum water volume at a minimum boron concentration, as presented in the Core Operating Limits Report, is required from the boric acid storage tanks or the refueling water storage tank.

The Boric Acid Tank and Refueling Water Storage Tank volumes required in Modes 5-6 to provide necessary SDM are based on the following inputs as discussed previously:

Boric Acid Tank

Required Volume for maintaining SDM	presented in the Core Operating Limits Report
Unusable Volume, Vortexing, Inst. Error	10,846 gallons
Additional Margin	569 gallons

Refueling Water Storage Tank

Required Volume for Maintaining SDM	presented in the Core Operating Limits Report
Water Below the Nozzle	13,442 gallons
Instrument Inaccuracy	11,307 gallons
Vortexing	13,247 gallons
Additional Margin	3,504 gallons

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

BORATION SYSTEMS (Continued)

The limits on contained water volume and boron concentration of the refueling water storage tank also ensure a pH value of between 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, and 120 steps and fully withdrawn (≥ 225 steps) for the Control Banks and 18 and 210 steps and fully withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs from the cold leg injection accumulators in the event the Reactor Coolant System pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The allowed outage time for the accumulators are variable based upon boron concentration to ensure that the reactor is shut down following a LOCA and that any problems are corrected in a timely manner. The minimum boron concentration required to ensure post-LOCA subcriticality, as presented in the Core Operating Limits Report, is based on nominal accumulator volume conditions and allows additional outage time since subcriticality is assured when the boron concentration is above this value. A slightly higher boron concentration, the minimum accumulator boron concentration limit for LCO 3.5.1c presented in the Core Operating Limits Report, is based on worst case liquid mass, boron concentration and measurement errors. A concentration less than this LCO value in any single accumulator or as a volume weighted average may be indicative of a problem, such as valve leakage. Since reactor shutdown is assured if the boron concentration is above the minimum concentration to ensure post-LOCA subcriticality and the accumulator volume is greater than or equal to the nominal volume, additional time is allowed to restore boron concentration in the accumulators.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

REFUELING WATER STORAGE TANK (Continued)

The Technical Specification Volume 363,513 gallons was determined by correcting the tank's low level setpoint (level at which makeup is added to tank) for instrument inaccuracy. This level provides the maximum available volume to account for shutdown margin, worst case single failure, adequate containment sump volume for transfer to recirculation, and sufficient volume above the switchover initiation level such that no operator action is required prior to ten minutes after the initiation of the accident.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the refueling water storage tank also ensure a pH value of between 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC in accordance with 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference Limits, target band*, and APL^{ND}* for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), W(Z)**, APL^{ND}**, $F_Q^L(X,Y,Z)$ and $W(Z)_{BL}$** for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, $F\Delta H^{L***}(X,Y)$ or, $F_{\Delta H}^{RTP****}$, and Power Factor Multiplier, $MF\Delta H$****, limits for Specification 3/4.2.3.
7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 2.2.1.
8. Boric Acid Storage System and Refueling Water Storage Tank volume and boron concentration limits for Specifications 3/4.1.2.5 and 3/4.1.2.6.
9. Reactor Water Makeup Pump flowrate limits for Specifications 3/4.3.3.12 and 3/4.9.2.

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* Reference 5 is not applicable to target band and APL^{ND}.

** References 4 and 5 are not applicable to W(Z), and APL^{ND}, and $W(Z_{BL})$.

*** Reference 1 is not applicable to $F\Delta H^L$.

**** Reference 5 is not applicable to $F_{\Delta H}^{RTP}$ and $MF_{\Delta H}$.

CORE OPERATING LIMITS REPORT (Continued)

10. Accumulator and Refueling Water Storage Tank boron concentration limits for Specifications 3/4.5.1 and 3/4.5.4.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 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.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).

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

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

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

5. DPC-NE-2011P-A, "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 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.)

6. DPC-NE-3001P-A, "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.)

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.)

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.