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

DUKE POWER COMPANY

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 143
License No. NPF-9

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Facility Operating License No. NPF-9 filed by the Duke Power Company (licensee) dated January 13, 1993, as supplemented January 28, February 17 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-9 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 143 , are hereby incorporated into this license. The licensee 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

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: May 31, 1994

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

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. NPF-17

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Facility Operating License No. NPF-17 filed by the Duke Power Company (licensee) dated January 13, 1993, as supplemented January 28, February 17 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-17 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 125 , are hereby incorporated into this license. The licensee 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

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: May 31, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 143

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 125

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

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
2-8	2-8
2-9	2-9
2-10	2-10
2-11	2-11
3/4 1-11	3/4 1-11
3/4 1-11a	-
3/4 1-12	3/4 1-12
3/4 1-12a	-
3/4 5-1	3/4 5-1
3/4 5-1a	-
3/4 5-2	3/4 5-2
3/4 5-2a	-
3/4 5-12	3/4 5-12
3/4 5-12a	-
B 3/4 1-2	B 3/4 1-2
B 3/4 1-2a	-
B 3/4 1-3	B 3/4 1-3
B 3/4 1-3a	-
B 3/4 1-4	B 3/4 1-4*
B 3/4 1-5	B 3/4 1-5*
B 3/4 5-1	B 3/4 5-1
B 3/4 5-1a	-
B 3/4 5-3	B 3/4 5-3
6-21	6-21
6-21a	6-21a
-	6-21b

*overflow pages - no changes made

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$(\Delta T/\Delta T_0)\ \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right)\left(\frac{1}{1 + \tau_3 S}\right) \leq K_1 - K_2\left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right)\left[T\left(\frac{1}{1 + \tau_6 S}\right) - T_1\right] + K_3\,(P - P^1) - f_1(\Delta I)$$

Where:

ΔT	=	Measured ΔT by Loop Narrow Range RTD,
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER,
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT,
τ_1, τ_2	=	Time constants utilized in the lead-lag controller for ΔT, as presented in the Core Operating Limits Report,
$\frac{1}{1 + \tau_3}$	=	Lag compensator on measured ΔT,
τ_3	=	Time constant utilized in the lag compensator for ΔT, as presented in the Core Operating Limits Report,
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 controller for T_{avg} dynamic compensation,
τ_4, τ_5	=	Time constants utilized in the lead-lag controller 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},

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the Core Operating Limits Report,

T' = $\leq$ 588.2 °F Reference T_{avg} at RATED THERMAL POWER,

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, sec^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear 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 imbalance 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 imbalance 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" slop presented in the Core Operating Limits Report.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: OVERPOWER ΔT

$$(\Delta T/\Delta T_0)\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)\left(\frac{1}{1+\tau_3 S}\right) \le K_4 - K_5\left(\frac{\tau_7 S}{1+\tau_7 S}\right)\left(\frac{1}{1+\tau_6 S}\right)T - K_6[T(\frac{1}{1+\tau_6 S} - T''] - f_2(\Delta I)$$

Where: ΔT = As defined in Note 1,

ΔT_o = 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,

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,

K_6 = Overpower ΔT reactor trip heatup setpoint penalty coefficient as presented in the Core Operating Limits Report for $T > T''$ and $K_6 = 0$ for $T \le T''$,

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

T = As defined in Note 1,

T'' = ≤ 588.2 °F Reference T_{avg} at RATED THERMAL POWER,

S = As defined in Note 1, and

$f_2(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear 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 imbalance 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 imbalance that the 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 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.6% of Rated Thermal Power.

Note 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.2% 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 and at least one associated Heat Tracing 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 RWST temperature when it is the source of borated water and the outside air temperature is less than 70°F.

REACTIVITY CONTROL SYSTEMS

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 and at least one associated Heat Tracing 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 3.5.5a, 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 storage 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.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 6870 and 7342 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 639 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one 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 reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

b. With one 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 reduce Reactor Coolant System pressure to less than 1000 psig 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.

   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

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*Reactor Coolant System pressure above 1000 psig.

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

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

   1) Verifying 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 1% of tank volume not resulting from normal makeup by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that power to the isolation valve operator is disconnected; and

d. At least once per 18 months by verifying proper operation of the power disconnect circuit.

4.5.1.1.2 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.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of at least 372,100 gallons,

b. A boron concentration 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 RWST 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

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2) Verifying the boron concentration of the water.

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

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated Heat Tracing Systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN 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. The minimum borated water volumes and concentrations required to maintain shutdown margin for the Boric Acid Storage System and the Refueling Water Storage Tank are presented in the Core Operating Limits Report.

The Technical Specification LCO value for the Boric Acid Storage Tank and the Refueling Water Storage Tank minimum contained water volume during Modes 1-4 is based on the required volume to maintain shutdown margin, an allowance for unusable volume and additional margin as follows:

Boric Acid Storage Tank Requirements for Maintaining SDM - Modes 1-4

Required volume for maintaining SDM	presented in the COLR
Unusable volume (to maintain full suction pipe)	4,132 gallons
Additional margin	6,470 gallons

BORATION SYSTEMS (Continued)

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

Required volume for maintaining SDM	presented in the COLR
Unusable volume (below nozzle)	16,000 gallons
Additional margin	17,893 gallons

With the RCS temperature below 200°F, one Boron Injection System 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 System becomes 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 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

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. The minimum borated water volumes and concentrations required to maintain shutdown margin for the Boric Acid Storage System and the Refueling Water Storage Tank are presented in the Core Operating Limits Report.

The Technical Specification LCO value for the Boric Acid Storage Tank and the Refueling Water Storage Tank minimum contained water volume during Modes 5 and 6 is based on the required volume to maintain shutdown margin, an allowance for unusable volume and additional margin as follows:

Boric Acid Storage Tank Requirements for Maintaining SDM - Modes 5 & 6

Required volume for maintaining SDM	presented in the COLR
Unusable volume (to maintain full suction pipe)	4,132 gallons
Additional margin	1,415 gallons

Refueling Water Storage Tank Requirements for Maintaining SDM - Modes 5 & 6

Required volume for maintaining SDM	presented in the CORL
Unusable volume (below nozzle)	16,000 gallons
Additional margin	6,500 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 RWST 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.

The control rod insertion limit and shutdown rod insertion limits are specified in the CORE OPERATING LIMITS REPORT per specification 6.9.1.9.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

For Specification 3.1.3.1 ACTIONS c. and d., it is incumbent upon the plant personnel to verify the trippability of the inoperable control rod(s). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism.

MOVABLE CONTROL ASSEMBLIES (Continued)

During performance of the Control Rod Movement periodic test (Specification 4.1.3.1.2), there have been some "Control Malfunctions" that prohibited a control rod bank or group from moving when selected, as evidenced by the demand counters and DRPI*. In all cases, when the control malfunctions were corrected, the rods moved freely (no excessive friction or mechanical interference) and were trippable.

This surveillance test is an indirect method of verifying the control rods are not immovable or untrippable. It is highly unlikely that a complete control rod bank or bank group is immovable or untrippable. Past surveillance and operating history provide evidence of "trippability."

Based on the above information, during performance of the rod movement test, if a complete control rod bank or group fails to move when selected and can be attributed to a "Control Malfunction," the control rods can be considered "Operable" and plant operation may continue while ACTIONS c. and d. are taken.

If one or more control rods fail to move during testing (not a complete bank or group and cannot be contributed to a "Control Malfunction"), the affected control rod(s) shall be declared "Inoperable" and ACTION a. taken.

(Reference: W letter dated December 21, 1984, NS-NRC-84-2990, E. P. Rahe to Dr. C. O. Thomas)

*Digital Rod Position Indicators

B.SES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) Cold Leg Accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS 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)

for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

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 RWST 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.

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}** and $W(Z)_{BL}$** for Specification 3/4.2.2, and

6. Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{L}$*** , 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. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3/4.5.1 and 3/4.5.5.

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.)

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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), 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

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 FQ Methodology.)

3. WCAP-10265-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-10168P, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," September 1989 (B&W Proprietary).

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

5. DPC-NE-2011P, "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, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," March 1991 (DPC Proprietary).

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

7. DPC-NE-2010P, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," April 1984 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

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

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

CORE OPERATING LIMITS REPORT

9. DPC-NE-3000, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," May 1989.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Nuclear 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 Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.