March 6, 1992

Docket Nos. 50-369
and 50-370

Distribution
See next page

Mr. T. C. McMeekin
Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, North Carolina 28078-8985

Dear Mr. McMeekin:

SUBJECT: ISSUANCE OF AMENDMENTS - McGUIRE NUCLEAR STATION, UNITS 1 AND 2 (TAC NO. M82340)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 130 to Facility Operating License NPF-9 and Amendment No. 112 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated December 18, 1991.

The amendments which support the McGuire Unit 2 Cycle 8 reload with B&W fuel, are administrative in nature and make McGuire Unit 2 TS identical to the previously reviewed and approved McGuire Unit 1 TS (November 27, 1991). The reload methodology and analyses that supported the previously approved Unit 1 TS amendments are conservatively bounding and applicable to both McGuire units.

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

Sincerely,

original signed by:
Timothy A. Reed, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 130to NPF-9
2. Amendment No. 112to NPF-17
3. Safety Evaluation

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

March 6, 1992

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and 50-370

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See next page

Mr. T. C. McMeekin
Duke Power Company

McGuire Nuclear Station

cc:

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Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

DATED: MARCH 6, 1992

AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NPF-9 - McGuire Nuclear Station, Unit 1

AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NPF-17 - McGuire Nuclear Station, Unit 2

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

DUKE POWER COMPANY

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
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 December 18, 1991, 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.

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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.130 , 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

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 6, 1992

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

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
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 December 18, 1991, 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:

   <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 112 , 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

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 6, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 130

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 112

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
Index	III	III
	V	V
	VI	VI
	XII	XII
	XVI	XVI
Appendix A	2-A1	2-1
	2-A2	2-2
	2-A4	2-4
	2-A5	2-5
	2-A6	2-6
	2-A7	2-7
	2-A8	2-8
	2-A9	2-9
	2-A10	2-10
	2-A11	2-11
	2-B1 - 2-B11	----
	3/4 A2-1	3/4 2-1
	3/4 A2-1a	3/4 2-1a
	3/4 A2-6	3/4 2-6
	3/4 A2-7	3/4 2-7
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	3/4 A2-15	3/4 2-15
	3/4 A2-15a	3/4 2-15a
	3/4 A2-19	3/4 2-19

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B B2-4	-----
B 3/4 A2-1	B 3/4 2-1
B 3/4 A2-2	B 3/4 2-2

(continued)

Remove Pages	Insert Pages
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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three loop operation, respectively.

APPLICABILITY: MODES 1 and 2

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

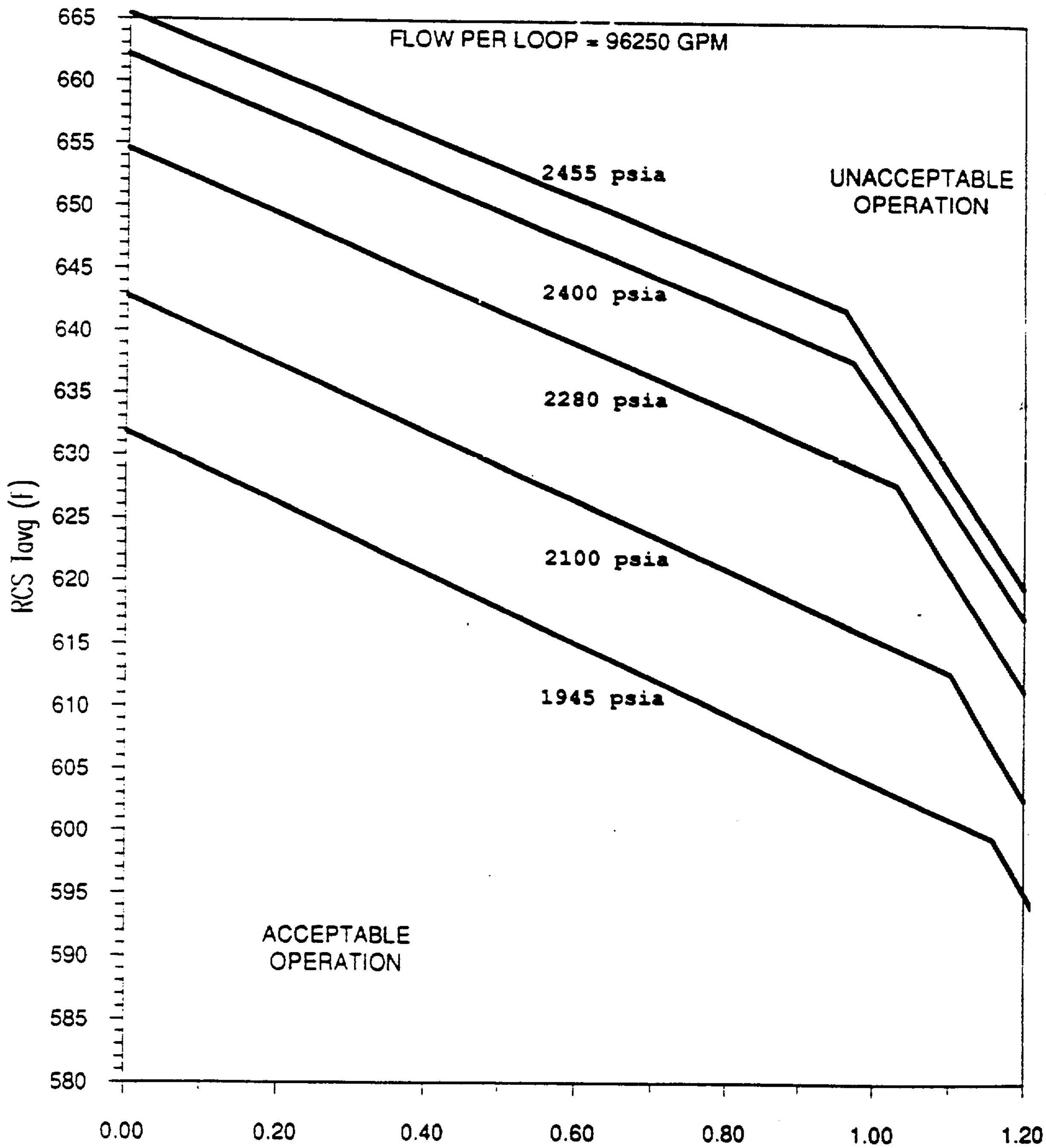
MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

Figure 2.1-1 Reactor Core Safety Limits -
Four Loops in Operation



Fraction of Rated Thermal Power

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlocks Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
5. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
6. Overtemperature ΔT	See Note 1	See Note 3
7. Overpower ΔT	See Note 2	See Note 4
8. Pressurizer Pressure--Low	$\geq$ 1945 psig	$\geq$ 1935 psig
9. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
10. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
11. Low Reactor Coolant Flow	$\geq$ 90% of minimum measured flow per loop*	$\geq$ 88.8% of minimum measured flow per loop*

*Minimum measured flow is 96,250 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
12. Steam Generator Water Level--Low-Low	$\geq$ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 40% of span at 100% of RATED THERMAL POWER	> 11% of span from 0 to 30% of RATED THERMAL POWER, increasing to 39.0% of span at 100% of RATED THERMAL POWER.
13. Undervoltage-Reactor Coolant Pumps	$\geq$ 5082 volts-each bus	$\geq$ 5016 volts-each bus
14. Underfrequency-Reactor Coolant Pumps	$\geq$ 56.4 Hz - each bus	$\geq$ 55.9 Hz - each bus
15. Turbine Trip		
a. Low Trip System Pressure	$\geq$ 45 psig	$\geq$ 42 psig
b. Turbine Stop Valve Closure	$\geq$ 1% open	$\geq$ 1% open
16. Safety Injection Input from ESF	N.A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7		
1) P-10 Input	10% of RATED THERMAL POWER	$\geq$ 9%, $\leq$ 11% of RATED THERMAL POWER
2) P-13 Input	$\leq$ 10% RTP Turbine Impulse Pressure Equivalent	$\leq$ 11% RTP Turbine Impulse Pressure Equivalent

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
c. Power Range Neutron Flux, P-8, Low Reactor Coolant Loop Flow, and Reactor Coolant Pump Breaker Position	≤ 48% of RATED THERMAL POWER	≤ 49% of RATED THERMAL POWER
d. Low Setpoint Power Range Neutron Flux, P-10, Enable Block of Source Intermediate and Power Range Reactor Trips	10% of RATED THERMAL POWER	≥ 9%, ≤ 11% of RATED THERMAL POWER
e. Turbine Impulse Chamber Pressure, P-13, Input to Low Power Reactor Trips Block P-7	≤ 10% RTP Turbine Impulse Pressure Equivalent	≤ 11% RTP Turbine Impulse Pressure Equivalent
18. Reactor Trip Breakers	N.A.	N.A.
19. Automatic Trip and Interlock Logic	N.A.	N.A.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT	=	Measured ΔT by Loop Narrow Range RTD
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT,
τ_1, τ_2	=	Time constants utilized in the lead-lag controller for ΔT, $\tau_1 \ge 8$ sec., $\tau_2 \le 3$ sec.,
$\frac{1}{1 + \tau_3}$	=	Lag compensator on measured ΔT,
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 <$ 2 sec.*
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER,
K_1	$\le$	1.1958,
K_2	=	0.03143
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	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}, $\tau_4 \ge 28$ sec, $\tau_5 \le 4$ sec.,
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, $\tau_6 < 2$ sec

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

K_3 = 0.001405,

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 -39% and +7.0%ΔI; $f_1(\Delta I) = 0$, 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 -39%ΔT, the ΔT Trip Setpoint shall be automatically reduced by 6.153% of ΔT_o, and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than +7.0%ΔI, the ΔT Trip Setpoint shall be automatically reduced by 1.511% of ΔT_o

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: OVERPOWER ΔT

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

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,
ΔT_o	=	As defined in Note 1,
K_4	$\le$	1.0809,
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}, $\tau_7 \ge 5$ sec,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,
K_6	=	0.001239/°F 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" = $\leq$ 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 -35% and +35% ΔI; $f_2(\Delta I) = 0$, 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 -35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_o; and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than +35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_o.

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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the acceptable limits as specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. For operation with the indicated AFD outside of the limits specified in the COLR,

   1. Either restore the indicated AFD to within the COLR limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitoring Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least two OPERABLE excore channels are indicating the AFD to be outside the limits.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(X,Y,Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(X,Y,Z)$ shall be limited by imposing the following relationship:

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

$$F_Q^{MA}(X,Y,Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5$$

Where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

$P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

$K(Z)$ = the normalized $F_Q(X,Y,Z)$ limit specified in the COLR for the appropriate fuel type, and

$F_Q^{MA}(X,Y,Z)$ = the measured heat flux hot channel factor $F_Q^M(X,Y,Z)$ with the adjustments specified in 4.2.2.3

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(X,Y,Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours, and

b. Control the AFD to within new AFD limits which are determined by reducing the allowable power at each point along the AFD limit lines of Specification 3.2.1 at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and reset the AFD alarm setpoints to the modified limits within 8 hours, and

c. POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit, and

d. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(X,Y,Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q^M(X,Y,Z)^{(1)}$ shall be evaluated to determine whether $F_Q(X,Y,Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Measuring $F_Q^M(X,Y,Z)$ at the earliest of:

1. At least once per 31 Effective Full Power Days, or

2. Upon reaching equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last determined[(2)], or

3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

---

(1) No additional uncertainties are required in the following equations for $F_Q^M(X,Y,Z)$ because the limits include uncertainties.

(2) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

c. Performing the following calculations:

1. For each core location, calculate the % margin to the maximum allowable design as follows:

$$\% \text{ Operational Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{OP}} \right) \times 100\%$$

$$\% \text{ RPS Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{RPS}} \right) \times 100\%$$

where $[F_Q^L(X,Y,Z)]^{OP}$ and $[F_Q^L(X,Y,Z)]^{RPS}$ are the Operational and RPS design peaking limits defined in the COLR.

2. Find the minimum Operational Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then either of the following actions shall be taken:

(a) Within 15 minutes:

(1) Control the AFD to within new AFD limits that are determined by:

$$(\text{AFD Limit})_{\substack{\text{reduced}\\ \text{negative}}} = (\text{AFD Limit})_{\substack{\text{COLR}^{(3)}\\ \text{negative}}} - \text{MARGIN}_{OP}^{MIN}$$

$$(\text{AFD Limit})_{\substack{\text{reduced}\\ \text{positive}}} = (\text{AFD Limit})_{\substack{\text{COLR}^{(3)}\\ \text{positive}}} - \text{MARGIN}_{OP}^{MIN}$$

where MARGIN_{OP}^{MIN} is the minimum margin from 4.2.2.2.c.1, and

(2) Within 8 hours, reset the AFD alarm setpoints to the modified limits of 4.2.2.2.c.2.a, or

(b) Comply with the ACTION requirements of Specification 3.2.2, treating the margin violation in 4.2.2.2.c.1 above as the amount by which F_Q^{MA} is exceeding its limit.

---

(3) Defined and specified in the COLR per Specification 6.9.1.9.

3. Find the minimum RPS Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then the following action shall be taken:

   Within 72 hours, reduce the K_1 value for OTΔT by:

   $$K_1^{\text{adjusted}} = K_1^{(4)} - [\text{KSLOPE}^{(3)} \times \text{Margin}^{\text{min}}_{\text{RPS}}]_{\text{absolute value}}$$

   where $\text{MARGIN}^{\text{min}}_{\text{RPS}}$ is the minimum margin from 4.2.2.2.c.1.

d. Extrapolating[5] at least two measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$$[F^M_Q(X,Y,Z)] \text{ (extrapolated)} \geq [F^L_Q(X,Y,Z)]^{OP} \text{ (extrapolated)}, \text{ and}$$

$$\frac{[F^M_Q(X,Y,Z)] \text{ (extrapolated)}}{[F^L_Q(X,Y,Z)]^{OP} \text{ (extrapolated)}} > \frac{[F^M_Q(X,Y,Z)]}{[F^L_Q(X,Y,Z)]^{OP}}$$

or

$$[F^M_Q(X,Y,Z)] \text{ (extrapolated)} \geq [F^L_Q(X,Y,Z)]^{RPS} \text{ (extrapolated)}, \text{ and}$$

$$\frac{[F^M_Q(X,Y,Z)] \text{ (extrapolated)}}{[F^L_Q(X,Y,Z)]^{RPS} \text{ (extrapolated)}} > \frac{[F^M_Q(X,Y,Z)]}{[F^L_Q(X,Y,Z)]^{RPS}}$$

either of the following actions shall be taken:

1. $F^M_Q(X,Y,Z)$ shall be increased by 2 percent over that specified in 4.2.2.2.a, and the calculations of 4.2.2.2.c repeated, or

---

(3) Defined and specified in the COLR per Specification 6.9.1.9.

(4) K_1 value from Table 2.2-1.

(5) Extrapolation of F^M_Q for the initial flux map taken after reaching equilibrium conditions is not required since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of F^M_Q 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.

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.2.2.c.1 shall be performed no later than the time at which the margin in 4.2.2.2.c.1 is extrapolated to be equal to zero.

e. The limits in Specifications 4.2.2.2.c and 4.2.2.2.d are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 When a full core power distribution map is taken for reasons other than meeting the requirements of Specification 4.2.2.2, an overall $F_Q^M(X,Y,Z)$ shall be determined, then increased by 3% to account for manufacturing tolerances, further increased by 5% to account for measurement uncertainty, and further increased by the radial-local peaking factor to obtain a maximum local peak. This value shall be compared to the limit in Specification 3.2.2.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}(X,Y)$ shall be limited by imposing the following relationship:

$$F^{M}_{\Delta H}(X,Y) \leq [F^{L}_{\Delta H}(X,Y)]^{LCO}$$

where: $F^{M}_{\Delta H}(X,Y)$ - the measured radial peak.

$[F^{L}_{\Delta H}(X,Y)]^{LCO}$ - the maximum allowable radial peak as defined in Core Operating Limits Report (COLR).

APPLICABILITY: MODE 1.

ACTION:

With $F_{\Delta H}(X,Y)$ exceeding its limit:

a. Within 2 hours, reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH%[1] for each 1% that $F^{M}_{\Delta H}(X,Y)$ exceeds the limit, and

b. Within 6 hours either:

   1. Restore $F^{M}_{\Delta H}(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

   2. Reduce the Power Range Neutron Flux-High Trip Setpoint in Table 2.2-1 at least RRH% for each 1% that $F^{M}_{\Delta H}(X,Y)$ exceeds that limit, and

c. Within 72 hours of initially being outside the limit of Specification 3.2.3, either:

   1. Restore $F^{M}_{\Delta H}(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

   2. Perform the following actions:

---

(1) RRH is the amount of THERMAL POWER reduction required to compensate for each 1% that $F^{M}_{\Delta H}(X,Y)$ exceeds the limit of Specification 3.2.3, provided in the COLR per Specification 6.9.1.9.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

ACTION:

(a) Reduce the OTΔT K_1 term in Table 2.2-1 by at least TRH[(2)] for each 1% that $F^M_{\Delta H}(X,Y)$ exceeds the limit, and

(b) Verify through incore mapping that $F^M_{\Delta H}(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

(2) TRH is the amount of OTΔT K_1 setpoint reduction required to compensate for each 1% that $F^M_{\Delta H}(X,Y)$ exceeds the limit of Specification 3.2.3, provided in the COLR per Specification 6.9.1.9.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

d. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a. and/or c.2 above; subsequent POWER OPERATION may proceed provided that $F^M_{\Delta H}(X,Y)$ is demonstrated, through incore flux mapping, to be within the Limit specified in the COLR prior to exceeding the following THERMAL POWER levels:

1. 50% of RATED THERMAL POWER,
2. 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^M_{\Delta H}(X,Y)$ shall be evaluated to determine whether $F_{\Delta H}(X,Y)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Measuring $F^M_{\Delta H}(X,Y)$ according to the following schedule:

1. Upon reaching equilibrium conditions after exceeding 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F^M_{\Delta H}(X,Y)$ was last determined[(3)], or
2. At least once per 31 Effective Full Power Days, or
3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

c. Performing the following calculations:

1. For each location, calculate the % margin to the maximum allowable design as follows:

(3) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

$$\% \ F_{\Delta H} \text{ Margin} = \left(1 - \frac{F^M_{\Delta H}(X,Y)}{[F^L_{\Delta H}(X,Y)]^{surv}}\right) \times 100\%$$

No additional uncertainties are required for $F^M_{\Delta H}(X,Y)$, because $[F^L_{\Delta H}(X,Y)]^{surv}$ includes uncertainties.

2. Find the minimum margin of all locations examined in 4.2.3.2.c.1 above. If any margin is less than zero, comply with the ACTION requirements of Specification 3.2.3 as if $[F^L_{\Delta H}(X,Y)]^{surv}$ is the same as $[F^L_{\Delta H}(X,Y)]^{LCO}$.

d. Extrapolating[(4)] at least two measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$F^M_{\Delta H}(X,Y)$ (extrapolated) $\geq [F^L_{\Delta H}(X,Y)]^{surv}$ (extrapolated) and

$$\frac{F^M_{\Delta H}(X,Y) \quad \text{(extrapolated)}}{[F^L_{\Delta H}(X,Y)]^{surv} \text{ (extrapolated)}} > \frac{F^M_{\Delta H}(X,Y)}{[F^L_{\Delta H}(X,Y)]^{surv}}$$

either of the following actions shall be taken:

1. $F^M_{\Delta H}(X,Y)$ shall be increased by 2 percent over that specified in 4.2.3.2.a, and the calculations of 4.2.3.2.c repeated, or

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.3.2.c shall be performed no later than the time at which the margin in 4.2.3.2.c is extrapolated to be equal to zero.

(4) Extrapolation of $F^M_{\Delta H}$ for the initial flux map taken after reaching equilibrium conditions is not required since the initial flux map establishes the baseline measurement for future trending.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.**

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Within 2 hours either:

      a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.
**Not applicable until calibration of the excore detectors is completed subsequent to refueling.

ACTION: (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02, within 30 minutes;

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1.

a. Reactor Coolant System T_{avg},

b. Pressurizer Pressure, and

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1.

ACTION:

a. With either of the parameters identified in 3.2.5a. and b. above exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

b. With the combination of Reactor Coolant System total flow rate and THERMAL POWER within the region of restricted operation specified on Figure 3.2.1, within 6 hours reduce the Power Range Neutron Flux-High Trip Setpoint to below the nominal setpoint by the same amount (% RTP) as the power reduction required by Figure 3.2-1.

c. With the combination of RCS total flow rate and THERMAL POWER within the region of prohibited operation specified on Figure 3.2-1:

1. Within 2 hours either:

a) Restore the combination of RCS total flow rate and THERMAL POWER to within the region of permissible operation,

b) Restore the combination of Reactor Coolant System total flow rate and THERMAL POWER to within the region of restricted operation and comply with action b. above, or

c) Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

2. Within 24 hours of initially being within the region of prohibited operation specified in Figure 3.2-1, verify that the combination of THERMAL POWER and RCS total flow rate are restored to within the regions of permissible or restricted operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

3/4.2.5 DNB PARAMETERS

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be measured by averaging the indications (meter or computer) of the operable channels and verified to be within their limits at least once per 12 hours.

4.2.5.2 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

TABLE 3.2-1

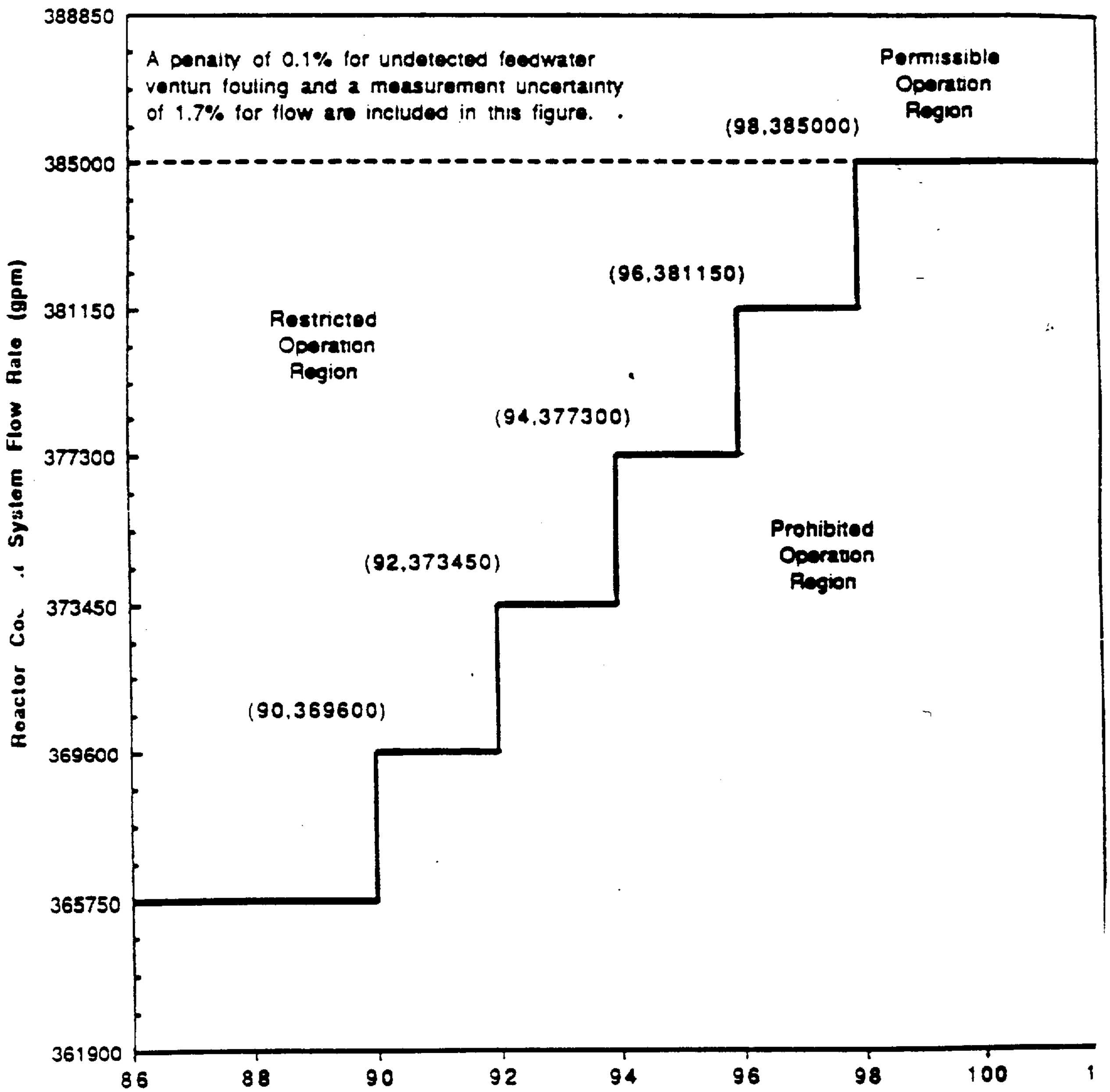
DNB PARAMETERS

PARAMETER	INDICATION	# OPERABLE CHANNELS	LIMITS*
Indicated Reactor Coolant System T_{avg}	meter	4	≤590.5°F
	meter	3	≤590.2°F
	computer	4	≤591.0°F
	computer	3	≤590.8°F
Indicated Pressurizer Pressure**	meter	4	≥2226.5 psig
	meter	3	≥2229.8 psig
	computer	4	≥2221.7 psig
	computer	3	≥2224.2 psig
Reactor Coolant System Total Flow Rate			Figure 3.2-1

*Limits applicable during four-loop operation.

**Limits not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

Figure 3.2 - 1. Reactor Coolant System Total Flow Rate Versus Rated Thermal Power - Four Loops in Operation



Fraction of Rated Thermal Power

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 25 - With one of the two trains of doghouse water level instrumentation inoperable (less than the minimum required number of channels operable), restore the inoperable train to operable status in 72 hours. After 72 hours with one train inoperable, or within one hour with 2 trains inoperable, monitor doghouse water level in the affected doghouse continuously until both trains are restored to operable status.

ACTION 26 - With any of the eight channels inoperable, place the inoperable channel(s) in the start permissive mode within one hour and apply the applicable action statement (Containment Spray - T.S. 3.6.2, Containment Air Return/Hydrogen Skimmer - T.S. 3.6.5.6).

ACTION 27 - With the number of OPERABLE channels one less than the total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System Instrumentation channels and interlocks of Table 3.3-1 (Unit 1) shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System Instrumentation channel and interlock shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.3 The response time of RTDs associated with the Reactor Trip System shall be demonstrated to be within their limits (see note 2 to Table 3.3-2) at least once per 18 months.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux - High Setpoint	4	2	3	1, 2	2
Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
5. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
c. Shutdown	2	0	1	3, 4, and 5	5
6. Overtemperature ΔT					
Four Loop Operation	4	2	3	1, 2	6
Three Loop Operation	(**)	(**)	(**)	(**)	(**)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Overpower WT					
Four Loop Operation	4	2	3	1, 2	6
Three Loop Operation	(**)	(**)	(**)	(**)	(**)
8. Pressurizer Pressure-Low	4	2	3	1	6 (***)
9. Pressurizer Pressure--High	4	2	3	1, 2	6 (***)
10. Pressurizer Water Level--High	3	2	2	1	6
11. Low Reactor Coolant Flow					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6
12. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6 (***)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
13. Undervoltage-Reactor Coolant Pumps (above P-7)	4-1/bus	2	3	1	6
14. Underfrequency-Reactor Coolant Pumps (above P-7)	4-1/bus	2	3	1	6
15. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1	6
b. Turbine Stop Valve Closure	4	4	1	1	11
16. Safety Injection Input from ESF	2	1	2	1, 2	9
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Low Setpoint Power Range Neutron Flux, P-10	4	2	3	1, 2	8
e. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10
19. Automatic Trip and Interlock Logic	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the Reactor Trip System breakers in the closed position, the Control Rod Drive System capable of rod withdrawal.

**Values left blank pending NRC approval of three loop operation.

***Comply with the provisions of Specification 3.3.2 for any portion of the channel required to be OPERABLE by Specification 3.3.2.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1 and Specification 4.3.2.1.

ACTION 7- Delete

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 12 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 9. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT		RESPONSE TIME
1.	Manual Reactor Trip	N.A.
2.	Power Range, Neutron Flux	≤0.5 second (1)
3.	Power Range, Neutron Flux, High Positive Rate	N.A.
4.	Intermediate Range, Neutron Flux	N.A.
5.	Source Range, Neutron Flux	N.A.
6.	Overtemperature ΔT	≤10.0 seconds (1)(2)
7.	Overpower ΔT	≤10.0 seconds (1)(2)
8.	Pressurizer Pressure--Low	≤2.0 seconds
9.	Pressurizer Pressure--High	≤2.0 seconds
10.	Pressurizer Water Level--High	N.A.

(1) Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.
(2) The ≤ 10.0 second response time includes a 6.5 second delay for the RTDs mounted in thermowells.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	$\leq$1.0 second
b. Two Loops (Above P-7 and below P-8)	$\leq$1.0 second
12. Steam Generator Water Level--Low-Low	$\leq$2.0 (Unit 1), 3.5 (Unit 2) seconds
13. Undervoltage-Reactor Coolant Pumps	<1.5 seconds
14. Underfrequency-Reactor Coolant Pumps	<0.6 second
15. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
16. Safety Injection Input from ESF	N.A.
17. Reactor Trip System Interlocks	N.A.
18. Reactor Trip Breakers	N.A.
19. Automatic Trip and Interlock Logic	N.A.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R (11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	M	N.A.	N.A.	1, 2
Low Setpoint	S	R(4)	M	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
4. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	1###, 2
5. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),M(9)	N.A.	N.A.	2##, 3, 4, 5
6. Overtemperature ΔT	S	R	M	N.A.	N.A.	1, 2
7. Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
8. Pressurizer Pressure--Low	S	R	M	N.A.	N.A.	1
9. Pressurizer Pressure--High	S	R	M	N.A.	N.A.	1, 2
10. Pressurizer Water Level--High	S	R	M	N.A.	N.A.	1
11. Low Reactor Coolant Flow	S	R	M	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
12. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	1, 2
13. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
14. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
15. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
16. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	M	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	M (8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	M (8)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
d. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	M (8)	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M (8)	N.A.	N.A.	1
18. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 12)	N.A.	1, 2, 3*, 4*, 5*
19. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*
20. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	M (13), R (14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With the Reactor Trip System breakers closed and the Control Rod Drive System capable of rod withdrawal.

- Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

- Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) - If not performed in previous 7 days.

(2) - Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) - Single point comparison of incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - Detector plateau curves shall be obtained, evaluated, and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) - Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) - Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) - With power greater than or equal to the interlock Setpoint the required operational test shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) - Monthly surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Monthly surveillance shall include verification of the High Flux at Shutdown Alarm Setpoint of less than or equal to five times background.

(10) - Setpoint verification is not required.

TABLE 4.3-1 (Continued)

TABLE NOTATION

(11) - The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function.

(12) - The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(13) - Prior to placing breaker in service, a local manual shunt trip shall be performed.

(14) - The automative undervoltage trip capability shall be verified operable.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) Instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation channel or interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the Trip Setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS Instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by the performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

##Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam pressure is not blocked.

**These values left blank pending NRC approval of three loop operation.

Note 1: Turbine driven auxiliary feedwater pump will not start on a blackout signal coincident with a safety injection signal.

ACTION STATEMENTS

ACTION 14 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 15a With the number of OPERABLE channels less than the total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour. With more than one channel inoperable, enter Specification 3.8.1.1.

ACTION 16 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.11 and Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the action required by Specification 3.7.1.4.

ACTION 24 - With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated auxiliary feedwater pump inoperable and take the action required by Specification 3.7.1.2. With the channels associated with more than one auxiliary feedwater pump inoperable, immediately declare the associated auxiliary feedwater pumps inoperable and take the action required by Specification 3.7.1.2.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Safety Injection, Reactor Trip, Feedwater Isolation, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water.		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High	$\leq$ 1.1 psig	$\leq$ 1.2 psig
d. Pressurizer Pressure--Low-Low	$\geq$ 1845 psig	$\geq$ 1835 psig
e. Steam Line Pressure - Low	$\geq$ 775 psig	$\geq$ 755 psig
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
b. Phase "B" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig
c. Purge and Exhaust Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig
d. Negative Steam Line Pressure Rate - High	$\leq$ 100 psi with a rate/lag function time constant $\geq$ 50 seconds	$\leq$ 120 psi with a rate/lag function time constant $\geq$ 50 seconds
e. Steam Line Pressure - Low	$\geq$ 775 psig	$\geq$ 755 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	$\leq$ 82% of narrow range instrument span each steam generator	$\leq$ 83% of narrow range instrument span each steam generator
c. Doghouse Water Level-High (Feedwater Isolation Only)	12"	13"
6. Containment Pressure Control System		
Start Permissive/Termination (SP/T)	0.3 $\leq$ SP/T $\leq$ 0.4 PSIG	0.25 $\leq$ SP/T $\leq$ 0.45 PSIG

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	$\geq$ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 40.0% of span at 100% of RATED THERMAL POWER.	$\geq$ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 39.0% of span at 100% of RATED THERMAL POWER.
2) Start Turbine-Driven Pumps	12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 40.0% of span at 100% of RATED THERMAL POWER.	$\geq$ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 39.0% of span at 100% of RATED THERMAL POWER.
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	$\geq$ 2 psig	$\geq$ 1 psig
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
f. Station Blackout - Start Motor-Driven Pumps and Turbine-Driven Pump (Note 1)	3464 ± 173 volts with a 8.5 ± 0.5 second time delay	$\geq$ 3200 volts
g. Trip of Main Feedwater Pumps - Start Motor-Driven Pumps	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Automatic Switchover to Recirculation		
RWST Level	$\geq$ 90 inches	$\geq$ 80 inches
9. Loss of Power		
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	3464 ± 173 volts with a 8.5 ± 0.5 second time delay	$\geq$ 3200 volts
10. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	$\leq$ 1955 psig	$\leq$ 1965 psig
b. T_{avg}, P-12	$\geq$ 553°F	$\geq$ 551°F
c. Reactor Trip, P-4	N.A.	N.A.
d. Steam Generator Level, P-14	See Item 5. above for all Trip Setpoints and Allowable Values.	

Note 1: The turbine driven pump will not start on a blackout signal coincident with a safety injection signal.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Containment Isolation	
Phase "A" Isolation	N.A.
Phase "B" Isolation	N.A.
Purge and Exhaust Isolation	N.A.
d. Steam Line Isolation	N.A.
e. Feedwater Isolation	N.A.
f. Auxiliary Feedwater	N.A.
g. Nuclear Service Water	N.A.
h. Component Cooling Water	N.A.
i. Reactor Trip (from SI)	N.A.
j. Start Diesel Generators	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27^{(1)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 12
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water	$\leq 65^{(3)}/76^{(4)}$
h. Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
i. Start Diesel Generators	≤ 11

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low-Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(3)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 12
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water System	$\leq 76^{(1)}/65^{(3)}$
h. Component Cooling Water	$\leq 76^{(1)}/65^{(3)}$
i. Start Diesel Generators	≤ 11
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12^{(3)}/22^{(4)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 12
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water	$\leq 65^{(3)}/76^{(4)}$
h. Steam Line Isolation	≤ 10
i. Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
j. Start Diesel Generators	≤ 11
5. Containment Pressure-High-High	
a. Containment Spray	≤ 45
b. Containment Isolation-Phase "B"	N.A.
c. Steam Line Isolation	≤ 10
6. Steam Generator Water Level-High-High	
a. Turbine Trip	N.A.
b. Feedwater Isolation	≤ 12

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Steam Generator Water Level LowLow	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-driven Auxiliary Feedwater Pumps	$\leq$ 60
8. Negative Steam Line Pressure Rate - High	
Steam Line Isolation	$\leq$ 10
9. Start Permissive	
Containment Pressure Control System	N.A.
10. Termination	
Containment Pressure Control System	N.A.
11. Auxiliary Feedwater Suction Pressure - Low	
Auxiliary Feedwater Pumps (Suction Supply Automatic Realignment)	$\leq$ 13
12. RWST Level	
Automatic Switchover to Recirculation	$\leq$ 60
13. Station Blackout	
a. Start Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Start Turbine-Driven Auxiliary Feedwater Pump (6)	$\leq$ 60
14. Trip of Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
15. Loss of Power	
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	$\leq$ 11

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps, Safety Injection and RHR pumps.

(2) Valves 1KC305B and 1KC315B for Unit 1 and Valves 2KC305B and 2KC315B for Unit 2 are exceptions to the response times listed in the table. The following response times in seconds are the required values for these valves for the initiating signal and function indicated:

2.d	$\leq 30^{(3)}/40^{(4)}$
3.d	$\leq 30^{(3)}$
4.d	$\leq 30^{(3)}/40^{(4)}$

(3) Diesel generator starting and sequence loading delays <u>not</u> included. Offsite power available. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps and Safety Injection pumps.

(4) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps and Safety Injection pumps.

(5) Response time for motor-driven auxiliary feedwater pumps on all Safety Injection signal shall be less than or equal to 60 seconds. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for auxiliary feedwater pumps.

(6) The turbine driven pump does not start on a blackout signal coincident with a safety injection signal.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.4 Each main steam line isolation valve (MSLIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1 - With one MSLIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5% of RATED THERMAL POWER within 2 hours.

MODES 2 and 3 - With one MSLIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided:

a. The isolation valve is maintained closed, and

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 Each MSLIV shall be demonstrated OPERABLE by verifying full closure within 8 seconds when tested pursuant to Specification 4.0.5.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the BWCMV correlation in this application). The correlation DNBR set such that there is a 95% probability with 95% confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and the CHF correlation are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The combined DNBR uncertainty is used to establish a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.50 and a reference cosine axial power shape with a peak of 1.55. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.50\ [1 + (1/RRH)\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER, and
RRH is given in the COLR.

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Positive Rate

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the calculated DNBR in the core at or above the design limit during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria are not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(X,Y,Z)$ Heat Flux Hot Channel Factor, is defined as the local heat flux on the surface of a fuel rod at core location X,Y,Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}(X,Y)$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along a rod at core location X,Y to the average rod power.

K(z) is defined as the normalized $F_Q(X,Y,Z)$ limit for a given core height.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) ensure that $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ limits specified in the CORE OPERATING LIMITS REPORT (COLR) are not exceeded during either normal operation or in the event of xenon redistribution following power changes. The AFD envelop specified in the COLR has been adjusted for measurement uncertainty.

AXIAL FLUX DIFFERENCE (Continued)

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed ΔI power operating space during normal power operation. These alarms are active when power is greater than 50% of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded, and (2) in the event of a LOCA the ECCS acceptance criteria are not exceeded. The peaking limits are specified in the CORE OPERATING LIMITS REPORT (COLR) per Specification 6.9.1.9.

The heat flux hot channel factor and nuclear enthalpy rise hot channel factor are each measurable, but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F_{\Delta H}(X,Y)$ will be maintained within its limits provided Conditions a. through d. above are maintained.

The limits on the nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, are specified in the COLR as Maximum Allowable Radial Peaking (MARP) limits, obtained by dividing the Maximum Allowable Total Peaking (MAP) limit by the axial peak [AXIAL(X,Y)] for location (X,Y). By definition, the Maximum Allowable Radial Peaking limits will result in a DNBR for the limiting transient tht is equivalent to the DNBR calculated with a design $F_{\Delta H}(X,Y)$ value of 1.50 and a limiting reference axial power shape. For transition cores, MARP limits may be applied to both MARK-BW and optimized fuel types provided allowances for differences in DNBR are accounted for in the generation of MARP limits. The MARP limits specified in the COLR include allowances for mixed core DNBR effects. The relaxation of $F_{\Delta H}(X,Y)$ as a function of THERMAL POWER allows for a change in the radial pwoer shape for all permissible control bank insertion limits. This relaxation is implemented by the application of the following factors:

$k = [1 + (1/RRH)\ (1 - P)]$

where k = power factor multiplier applied to the MAP limits
p = THERMAL POWER / RATED THERMAL POWER
RRH is given in the COLR

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The hot channel factor $F_Q^M(X,Y,Z)$, and the nuclear enthalpy rise hot channel factor, $F_{\Delta}^M H(X,Y)$, are measured periodically to verify that the core is operating as designed. $F_Q^M(X,Y,Z)$ and $F_{\Delta H}^M$ (X,Y) are compared to allowable limits to provide reasonable assurance that limiting criteria will not be exceeded for operation within the Technical Specification limits of Sections 2.2 (Limiting Safety Systems Settings), 3.1.3 (Movable Control Assemblies), 3.2.1 (Axial Flux Difference), and 3.2.4 (Quadrant Power Tilt Ratio). A peaking margin calculation is performed to provide the basis for decreasing the width of the AFD and f(ΔI) limits and for reducing THERMAL POWER.

When an $F_Q^M(X,Y,Z)$ measurement is obtained from a full-core map in accordance with surveillance requirements of Specification 4.2.2, no uncertainties are applied to the measured peak since a measurement uncertainty of 5.0% and a manufacturing tolerance of 3.0% are included in the peaking limit. When $F_Q^M(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2, the measured peak is increased by the radial-local peaking factor and appropriate allowances for measurement uncertainty and for manufacturing tolerances.

When an $F_{\Delta H}^M(X,Y)$ measurement is obtained from a full-core map, regardless of the reason, no uncertainties are applied to the measured peak since the required uncertainties are included in the peaking limit.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required provides DNB and linear heat generation rate protection with the x-y plane power tilts. The peaking increase that corresponds to a QUADRANT POWER TILT RATIO of 1.02 is included in the generation of the AFD limits.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q(X,Y,Z)$ is reinstated by reducing the power by 3% from RATED THERMAL POWER for each percent of tilt in excess of 2.0%.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a design limit DNBR throughout each analyzed transient. As noted on Figure 3.2-1, RCS flow rate and THERMAL POWER may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the power level is decreased) to ensure that the calculated DNBR will not be below the design DNBR value. The relationship defined on Figure 3.2-1 remains valid as long as the limits placed on the nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, in Specification 3.2.3 are maintained. The indicated T_{avg} values and the indicated pressurizer pressure values correspond to analytical limits of 592.6°F and 2220 psia respectively, with allowance for indication instrumentation measurement uncertainty. When RCS flow rate is measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-1 since a measurement error of 1.7% for RCS total flow rate has been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conversative

3/4.2.5 DNB PARAMETERS (Continued)

manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-1. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. if detected, action shall be taken before performing subsequent precision heat balance mesurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. Indication instrumentation measurement uncertainties are accounted for in the limits provided in Table 3.2-1.

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat; however, single failure considerations require that three loops be OPERABLE. Also, the uncontrolled bank withdrawal from zero power or subcritical assumes three reactor coolant loops in operation.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 300°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated December 18, 1991, the Duke Power Company (the licensee or DPC) submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes revise the McGuire Unit 2 TS to be identical to the current McGuire Unit 1 TS and are in support of the McGuire Unit 2 Cycle 8 fuel reload. Both the reload methodology and the accident analyses used to support the current McGuire Unit 2 fuel reload are identical to that used to support the previous McGuire Unit 1 TS amendment.

2.0 EVALUATION

Beginning with fuel cycle 8 for both McGuire Units, the fuel being reloaded into McGuire Nuclear Station is manufactured by the B&W Fuel Company (BWFC). McGuire Unit 1 began reloading with B&W fuel during the most recent refueling outage that ended in December 1991. McGuire Unit 2 began reloading with B&W fuel during the refueling outage which began in January 1992. The analytical methodology used to support reloading with B&W fuel has been developed by BWFC and DPC. More importantly, the reload analytical methodology has been reviewed and approved by the NRC staff in response to a series of BWFC and DPC topical reports made in support of the previous McGuire Unit 1 Cycle 8 reload. As a result of the Unit 1 TS amendment, it was necessary to have two separate TSs for each McGuire Unit during the interim period until Unit 2 transitioned to BWFC fuel. The licensee's current submittal is therefore making both McGuire Unit TSs identical by making the previously approved Unit 1 TS changes applicable to both McGuire Units.

The reload methodology which forms the basis for both the previous Unit 1 reload submittal and the current Unit 2 reload TS submittal is based primarily on DPC topicals identified below.

The methods and results for analyses of the Steam System Piping Failure, Rod Ejection, and Dropped RCCA/RCCA Bank transients are documented in DPC-NE-3001. The analysis of other transients than the three identified above are described

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in DPC-NE-3000. The basis for choosing the initial conditions and assumptions used in each of the analyzed transients is found in DPC-NE-3002. The NRC staff has reviewed and approved the above three topical reports and found them acceptable for referencing in both Catawba and McGuire Nuclear Station licensing submittals.

The loss-of-coolant analysis for both McGuire Units is documented in BAW-10174. This topical has been reviewed and approved by the NRC staff and is acceptable for referencing in licensing submittals for McGuire Nuclear Station.

The core physics parameters for McGuire Cycle 8 were generated using the methodology in approved Topical Report DPC-NE-2010. The reactor protection system limits were verified using the methodology in approved Topical Report DPC-NE-2011.

The hydraulic compatibility of Mark BW and Westinghouse OFA fuel is addressed in approved topical report BAW-10173.

The staff reviewed in detail the application of the methodology identified above to McGuire Unit 1 following the licensee's submittals dated June 26, September 16, and November 7, 1991. The staff reviewed the TS changes that stem from the new methodology and found those TS changes to be acceptable in its SER supporting TS amendments 128 (Unit 1) and 110 (Unit 2). The TS changes that support the current McGuire Unit 2 Cycle 8 reload TS application basically revise Unit 2 to be identical with the previously found acceptable Unit 1 TS pages.

The licensee states in their current submittal that the McGuire Unit 2 Cycle 8 reload TS amendment is an administrative TS change. The staff concurs with the licensee in this assessment, since the analysis, methodology, inputs, and assumptions used to support the previously approved McGuire Unit 1 reload are conservatively bounding for both McGuire Units and have been reviewed and approved by the NRC staff. As a result the staff concludes that the TS changes proposed in the licensee's submittal are acceptable and these changes will return both McGuire Units to an identical set of Technical Specifications.

3.0 STATE CONSULTATION

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

4.0 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 (57 FR 4486). 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.

5.0 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: T. A. Reed, PDII-3/NRR

Date: **March 6, 1992**