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. 128
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 (the licensee) dated June 26, 1991, as supplemented September 16, 1991 and November 7, 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-9 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 128 , are hereby incorporated into the 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: November 27, 1991

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. 110
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 2 (the facility) Facility Operating License No. NPF-17 filed by the Duke Power Company (the licensee) dated June 26, 1991, as supplemented September 16, 1991 and November 7, 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:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 110 , are hereby incorporated into the 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

for

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

Attachment:
Technical Specification
Changes

Date of Issuance: November 27, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 128

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 110

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	Index
	2-A1
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	2-A4
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	2-A6
	2-A7
	2-A8
	2-A9
	2-A10
	2-A11
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	2-B10
	2-B11
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	3/4 A2-1
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	B A2-1
	B A2-4
	B A2-4a
	B A2-5
	B B2-1
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	B 3/4 A2-2
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SECTION 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 2.1-1a and 2.1-2 for four and three loop operation, respectively.

APPLICABILITY: MODES 1 and 2. (Unit 1 only)

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. (Units 1 and 2)

ACTION:

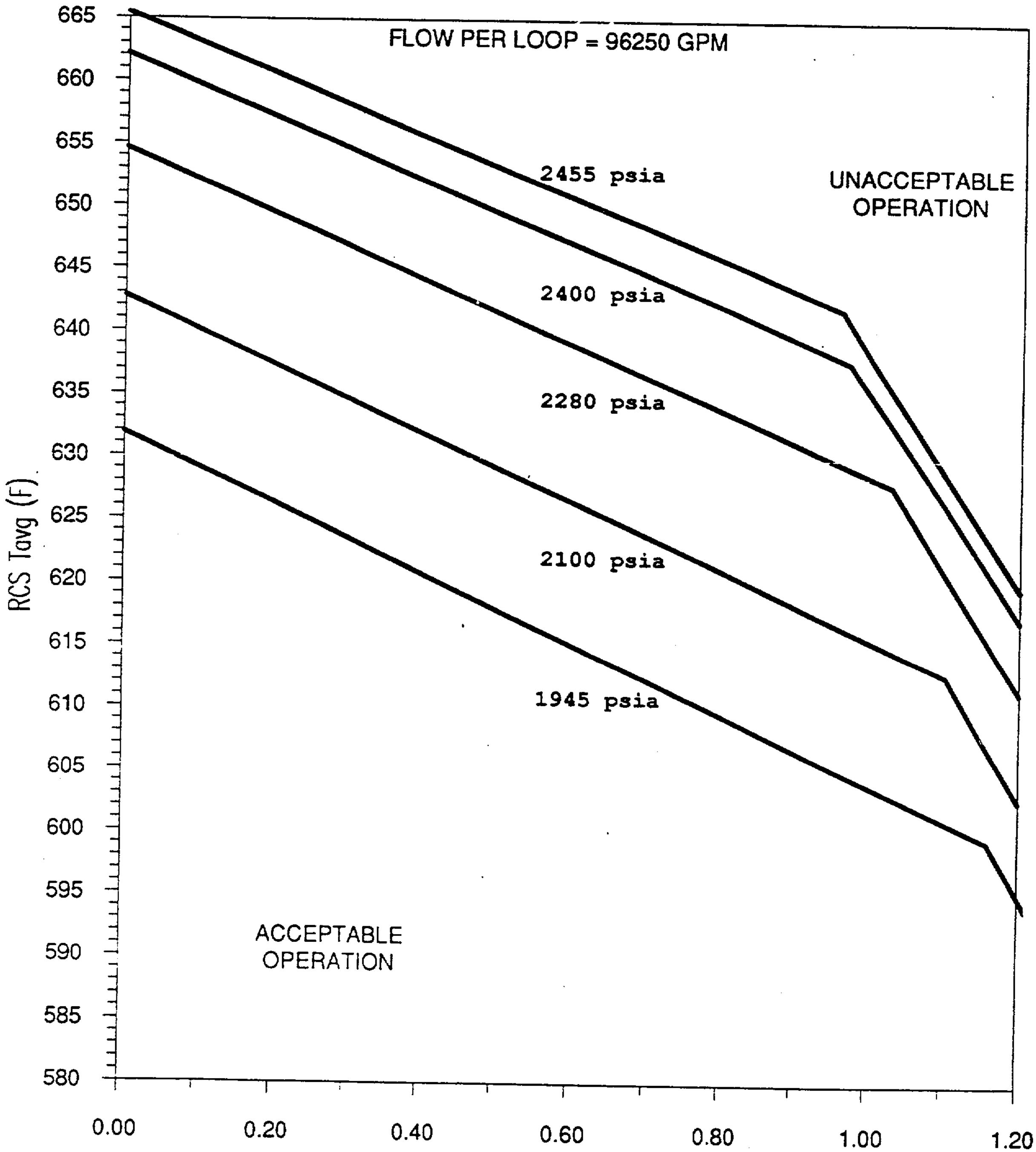
MODES 1 and 2

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 2.1-1a Reactor Core Safety Limits -
Four Loops in Operation (Unit 1)



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 2.2-1a (UNIT 1).

APPLICABILITY: As shown for each channel in Table 3.3-1a (UNIT 1).

ACTION:

With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1a, 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 2.2-1a

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 2.2-1a (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	$\geq$ 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 2.2-1a (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 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT	=	Measured ΔT by Loop Narrow Range 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, $\tau_1 \geq 8$ sec., $\tau_2 \leq 3$ sec.,
$\frac{1}{1 + \tau_3}$	=	Lag compensator on measured ΔT,
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 \leq$ 2 sec.*
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER,
K_1	$\leq$	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}, $\tau_4 \geq 28$ sec, $\tau_5 \leq 4$ sec.,
T	=	Average temperature, °F,
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg},

TABLE 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \leq 2$ 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

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

Where:

ΔT	=	As defined in Note 1,
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	As defined in Note 1
τ_1, τ_2	=	As defined in Note 1
$\frac{1}{1+\tau_3 S}$	=	As defined in Note 1,
ΔT_o	=	As defined in Note 1,
K_4	$\leq$	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}, $\tau_7 \geq 5$ 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 $T \leq T''$,

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

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.

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 2.1-1b and 2.1-2 for four and three loop operation, respectively.

APPLICABILITY: MODES 1 and 2. (Unit 2 only)

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. (Units 1 and 2)

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.

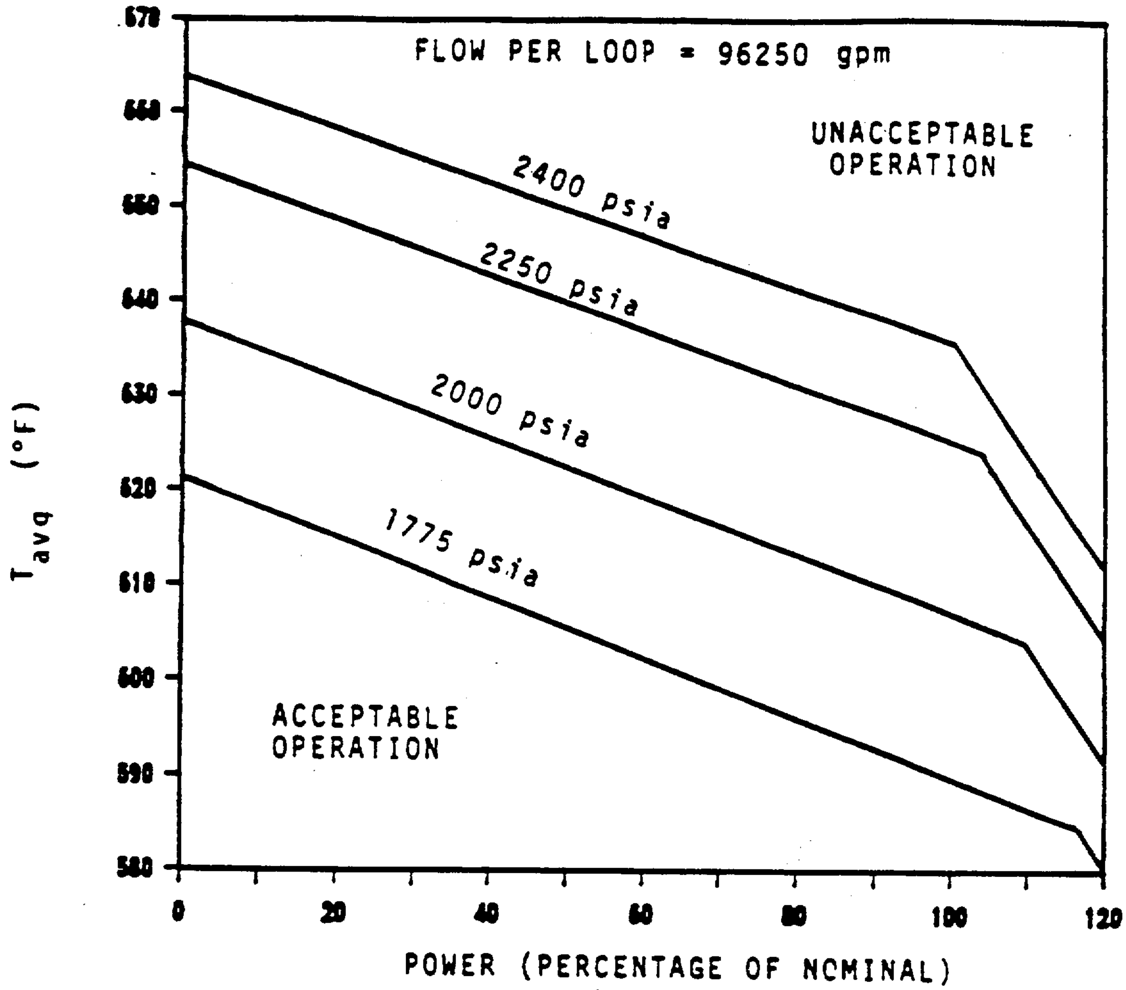


FIGURE 2.1-1b

UNIT 2

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

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 2.2-1b. (Unit 2)

APPLICABILITY: As shown for each channel in Table 3.3-1b. (Unit 2)

ACTION:

With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1b, 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 2.2-1b

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. Power Range, Neutron Flux, High Negative 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
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	$\geq$ 1945 psig	$\geq$ 1935 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. 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 2.2-1b (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

TABLE 2.2-1b (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	$\leq$ 48% of RATED THERMAL POWER	$\leq$ 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	$\geq$ 9%, $\leq$ 11% of RATED THERMAL POWER
e. Turbine Impulse Chamber Pressure, P-13, Input to Low Power Reactor Trips Block P-7	$\leq$ 10% RTP Turbine Impulse Pressure Equivalent	$\leq$ 11% RTP Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

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

Where: ΔT = Measured ΔT by Loop Narrow Range RTDs,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT,

τ_1, τ_2 = Time constants utilized in the lead-lag controller for ΔT, $\tau_1 \geq 8$ sec., $\tau_2 \leq 3$ sec.,

$\frac{1}{1 + \tau_3}$ = Lag compensator on measured ΔT,

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 \leq$ 2 sec.*

ΔT_o = Indicated ΔT at RATED THERMAL POWER,

K_1 $\leq$ 1.200,

K_2 = 0.0222

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = 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}, $\tau_4 \geq 28$ sec, $\tau_5 \leq 4$ sec.,

T = Average temperature, °F,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg},

TABLE 2.2-1b (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \leq 2$ sec
T'	=	$\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,
K_3	=	0.001095,
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 -29% and +7.0%; $f_1(\Delta I) = 0$, 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 that the magnitude of $q_t - q_b$ exceeds -29%, the ΔT Trip Setpoint shall be automatically reduced by 3.151% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +7.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.50% of its value at RATED THERMAL POWER.

TABLE 2.2-1b (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: OVERPOWER ΔT

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

Where:

ΔT	=	As defined in Note 1,
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	As defined in Note 1
τ_1, τ_2	=	As defined in Note 1
$\frac{1}{1 + \tau_3 S}$	=	As defined in Note 1,
ΔT_o	=	As defined in Note 1,
K_4	$\leq$	1.0900,
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}, $\tau_7 \geq 5$ sec,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,
K_6	=	0.00169/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

TABLE 2.2-1b (Continued) UNIT 2

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

$f_2(\Delta I)$ = 0 for all ΔI.

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 3a: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2%.

Note 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.2% of Rated Thermal Power.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misoperation

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

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*. (Unit 1)

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. (UNIT 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:

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

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

where $\text{MARGIN}_{\substack{\text{MIN}\\ \text{OP}}}$ 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:

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

   where MARGIN $_{\text{RPS}}^{\text{min}}$ 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:

$$[F_Q^M(X,Y,Z)] \text{ (extrapolated)} \geq [F_Q^L(X,Y,Z)]^{OP} \text{ (extrapolated), and}$$

$$\frac{[F_Q^M(X,Y,Z)] \text{ (extrapolated)}}{[F_Q^L(X,Y,Z)]^{OP} \text{ (extrapolated)}} > \frac{[F_Q^M(X,Y,Z)]}{[F_Q^L(X,Y,Z)]^{OP}}$$

or

$$[F_Q^M(X,Y,Z)] \text{ (extrapolated)} \geq [F_Q^L(X,Y,Z)]^{RPS} \text{ (extrapolated), and}$$

$$\frac{[F_Q^M(X,Y,Z)] \text{ (extrapolated)}}{[F_Q^L(X,Y,Z)]^{RPS} \text{ (extrapolated)}} > \frac{[F_Q^M(X,Y,Z)]}{[F_Q^L(X,Y,Z)]^{RPS}}$$

either of the following actions shall be taken:

1. $F_Q^M(X,Y,Z)$ 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_Q^M 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_Q^M 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.

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.

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. (UNIT 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.

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 $F^{M}_{\Delta H}(X,Y)$ 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

$$\frac{F^{M}_{\Delta H}(X,Y) \text{ (extrapolated)}}{[F^{L}_{\Delta H}(X,Y)]^{surv} \text{ (extrapolated)}} > \frac{F^{M}_{\Delta H}(X,Y)}{[F^{L}_{\Delta H}(X,Y)]^{surv}}$$

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*.** (Unit 1 only)

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.

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*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.

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 3.2-1a: (Unit 1)

a. Reactor Coolant System T_{avg},

b. Pressurizer Pressure, and

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1. (Unit 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 3.2-1a

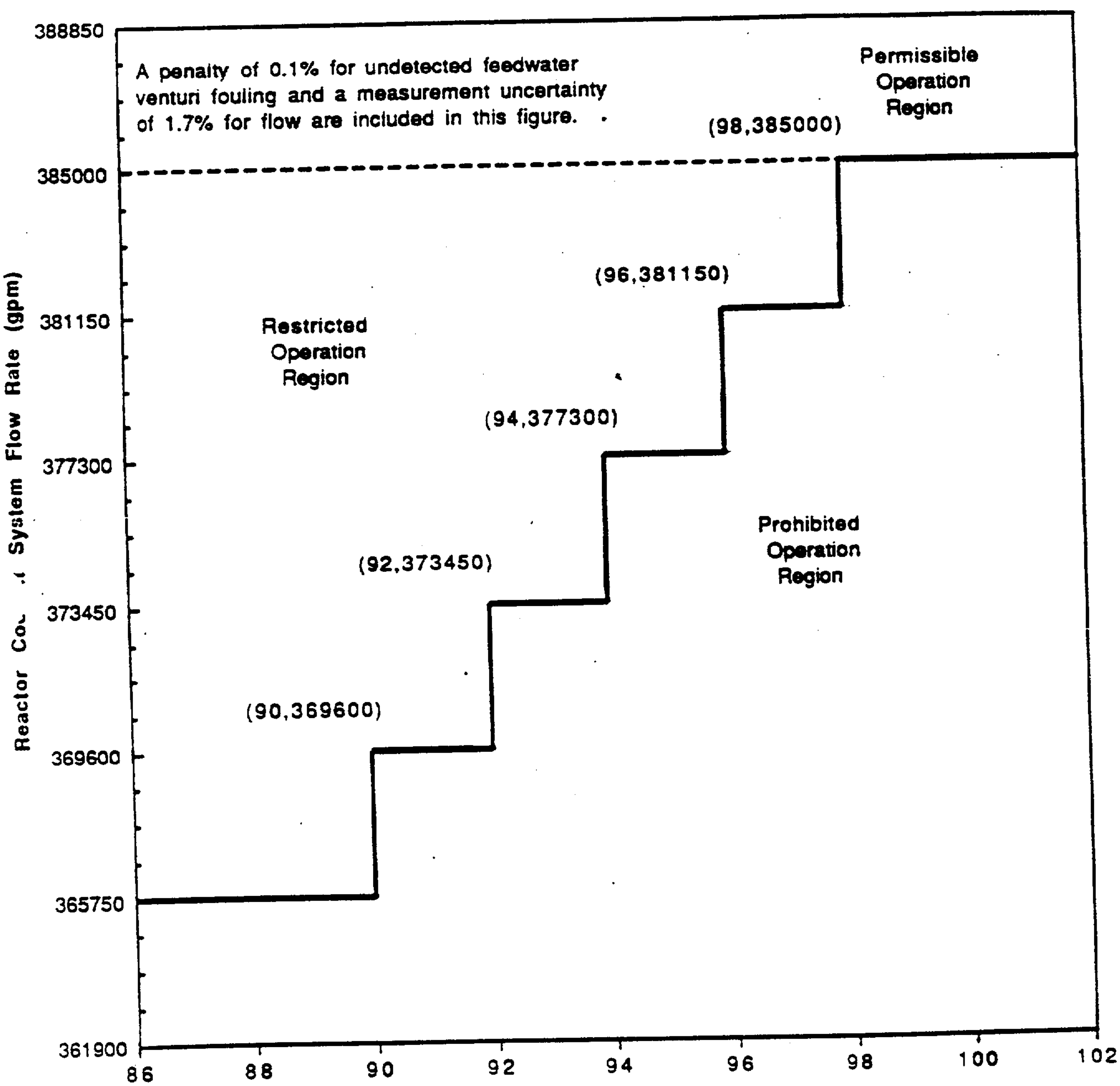
DNB PARAMETERS

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

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:

a. the allowed operational space as specified in the CORE OPERATING LIMITS REPORT (COLR) for RAOC operation, or

b. the target band specified in the COLR about the target flux difference during base load operation.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*. (Unit 2 only)

ACTION:

a. For RAOC 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. For base load operation above APL^{ND}** with the indicated AXIAL FLUX DIFFERENCE outside of the applicable target band about the target flux difference:

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

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

c. 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.

**APL^{ND} is the minimum allowable (nuclear design) power level for base load operation and is specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

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.

4.2.1.3 When in Base Load operation, the target axial flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference in conjunction with the surveillance requirements of Specification 3/4.2.2 or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationship:

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

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

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

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$, and

K(Z) = the normalized $F_Q(Z)$ for a given core height specified in the COLR.

APPLICABILITY: MODE 1. (Unit 2 only)

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit; and

b. 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 $F_Q(Z)$ 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 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ 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. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5%** to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{W(z) \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z), and W(z) are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or
2. At least once per 31 Effective Full Power Days, whichever occurs first.

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

**For Unit 1, Cycle 7, when the number of available moveable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 5% measurement uncertainty shall be increased to [5% + (3-T/14.5)(2%)] where T is the number of available thimbles.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c. or

2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

1) Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{P} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P \geq 0.5$$

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{0.5} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P < 0.5$$

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits of Specification 3.2.1 by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

c) Verify that the requirements of Specification 4.2.2.3 for base load operation are satisfied and enter base load operation.

g. The limits specified in Specifications 4.2.2.2c, 4.2.2.2e., and 4.2.2.2f. above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 Base load operation is permitted at powers above APL^{ND*} if the following conditions are satisfied:

a. Prior to entering base load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain base load operation surveillance (AFD within the target band about the target flux difference of Specification 3.2.1) during this time period. Base load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \underset{\text{over } Z}{\text{minimum}} \left[\frac{(F_Q^{RTP} \times K(Z))}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\%$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. F_Q^{RTP} is the F_Q limit. K(z) is the normalized $F_Q(z)$ as a function of core height. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation. F_Q^{RTP}, K(z), and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

b. During base load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering base load operation.

4.2.2.4 During base load operation $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5%** to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

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*APL^{ND} is the minimum allowable (nuclear design) power level for base load operation in Specification 3.2.1.

**For Unit 1, Cycle 7, when the number of available moveable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 5% measurement uncertainty shall be increased to [5% + (3-T/14.5)(2%)] where T is the number of available thimbles.

c. Satisfying the following relationship:

$$F_Q^M(Z) \le \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. F_Q^{RTP} is the F_Q limit. $K(Z)$ is the normalized $F_Q(Z)$ as a function of core height. P is the relative THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation. F_Q^{RTP}, $K(Z)$, and $W(Z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

d. Measuring $F_Q^M(Z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering base load operation after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and
2. At least once per 31 effective full power days.

e. With measurements indicating

$$\underset{\text{over } Z}{\text{maximum}} \left[\frac{F_Q^M(Z)}{K(Z)} \right]$$

has increased since the previous determination $F_Q^M(Z)$ either of the following actions shall be taken:

1. $F_Q^M(Z)$ shall be increased by 2 percent over that specified in 4.2.2.4.c, or
2. $F_Q^M(Z)$ shall be measured at least once per 7 EFPD until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(Z)}{K(Z)} \right] \text{ is not increasing.}$$

f. With the relationship specified in 4.2.2.4.c above not being satisfied, either of the following actions shall be taken:

1. Place the core in an equilbrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(Z)$, or

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated with the following expression:

$$[(\text{max. over z of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{F_Q^{RTP}}{P} \times K(Z)}]) -1] \times 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plan regions:

1. Lower core region 0 to 15 percent, inclusive.

2. Upper core region 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5%* to account for measurement uncertainty.

*For Unit 1, Cycle 7, when the number of available moveable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 5% measurement uncertainty shall be increased to [5% + (3-T/14.5)(2%)] where T is the number of available thimbles.

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation specified in the CORE OPERATING LIMITS REPORT (COLR) for four loop operation:

Where:

a. $R = \dfrac{F^N_{\Delta H}}{F^{RTP}_{\Delta H} \quad [1.0 + MF_{\Delta H}\ (1.0 - P)]}$,

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

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since the figure specified in the COLR includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 1.7% for flow and 4%* for incore measurement of $F^N_{\Delta H}$,

d. $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

e. $MF_{\Delta H}$ = The power factor multiplier specified in the COLR.

APPLICABILITY: MODE 1. (Unit 2)

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation specified in the COLR:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

2. 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.

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*For Unit 1, Cycle 7, when the number of available moveable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 4% measurement uncertainty shall be increased by changing the value of $F^{RTP}_{\Delta H}$ in the R equation to [(0.0149/14.5)T + 1.4453] where T is the number of available thimbles.

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. 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.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation specified in the COLR prior to exceeding the following THERMAL POWER levels:

   1. A nominal 50% of RATED THERMAL POWER,

   2. A nominal 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 The combination of indicated RCS total flow rate determined by process computer readings or digital voltmeter measurement and R shall be within the region of acceptable operation specified in the COLR:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation specified in the COLR at least once per 12 hours when the most recently obtained value of R obtained per Specification 4.2.3.2, is assumed to exist.

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

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

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*. (Unit 2 only)

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 1.0 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.

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 1.0, 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.

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.

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 3.2-1b (Unit 2):

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1 (Unit 2).

ACTION:

With any of the above parameters 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.

SURVEILLANCE REQUIREMENTS

4.2.5 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.

TABLE 3.2-1b

UNIT 2

DNB PARAMETERS

PARAMETER	INDICATION	# OPERABLE CHANNELS	LIMITS*
Indicated Reactor Coolant System T_{avg}	meter	4	$\leq$590.5°F
	meter	3	$\leq$590.2°F
	computer	4	$\leq$591.0°F
	computer	3	$\leq$590.8°F
Indicated Pressurizer Pressure**	meter	4	$\geq$2226.5 psig
	meter	3	$\geq$2229.8 psig
	computer	4	$\geq$2221.7 psig
	computer	3	$\geq$2224.2 psig

*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.

TABLE 3.3-1a

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 3.3-2a

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	$\leq$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	$\leq$10.0 seconds (1)(2)
7. Overpower ΔT	$\leq$10.0 seconds (1)(2)
8. Pressurizer Pressure--Low	$\leq$2.0 seconds
9. Pressurizer Pressure--High	$\leq$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 $\leq$ 10.0 second response time includes a 6.5 second delay for the RTDs mounted in thermowells.

TABLE 4.3-1a

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

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 3.3-4a (Unit 1) and with RESPONSE TIMES as shown in Table 3.3-5a (Unit 1).

APPLICABILITY: As shown in Table 3.3-3. (Unit 1 only)

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 3.3-4a

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 3.3-4a (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 3.3-4a (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 3.3-4a (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 3.3-4a (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 3.3-5a

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 3.3-5a (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 3.3-5a (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 3.3-1b

UNIT 2

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. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. 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
7. Overtemperature ΔT					
Four Loop Operation	4	2	3	1, 2	6
Three Loop Operation	(**)	(**)	(**)	(**)	(**)

TABLE 3.3-2b

UNIT 2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	$\leq$0.5 second (1)
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	$\leq$0.5 second (1)
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	$\leq$10.0 seconds (1)(2)
8. Overpower ΔT	$\leq$10.0 seconds (1)(2)
9. Pressurizer Pressure--Low	$\leq$2.0 seconds
10. Pressurizer Pressure--High	$\leq$2.0 seconds
11. 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 $\leq$ 10.0 second response time includes a 6.5 second delay for the RTDs mounted in thermowells.

TABLE 4.3-1b

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. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),M(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R	M	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	M	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R	M	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	M	N.A.	N.A.	1
12. Low Reactor Coolant Flow	S	R	M	N.A.	N.A.	1

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 3.3-4b (Unit 2) and with RESPONSE TIMES as shown in Table 3.3-5b (Unit 2).

APPLICABILITY: As shown in Table 3.3-3. (Unit 2 only)

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 3.3-4b

UNIT 2

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	$\geq$ 145 psig	$\geq$ 835 psig
e. Steam Line Pressure - Low	$\geq$ 585 psig	$\geq$ 565 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 3.3-4b (Continued)

UNIT 2
(SAME AS UNIT 1)

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 3.3-4b (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	$\geq$ 585 psig	$\geq$ 565 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 3.3-4b (Continued)

UNIT 2
(SAME AS UNIT 1)

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 3.3-4b (Continued)

(UNIT 2)
(SAME AS UNIT 1)

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 3.3-5b UNIT 2

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	≤ 9
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 3.3-5b (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	≤ 9
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	≤ 9
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	≤ 7
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	≤ 7
6. Steam Generator Water Level-High-High	
a. Turbine Trip	N.A.
b. Feedwater Isolation	≤ 9

TABLE 3.3-5b (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	$\leq$ 7
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

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE and in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILIYT: MODE 3

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than the above required reactor coolant loops in operation, restore the required loops to operation within 72 hours or open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 12% at least once per 12 hours.

4.4.1.2.2 At least the above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps may be de-energized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

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

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 585 and 639 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve or boron concentration less than 1900 ppm, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

c. With one accumulator inoperable due to boron concentration less than 1900 ppm and:

1) The volume weighted average boron concentration of the accumulators 1900 ppm or greater, restore the inoperable accumulator to OPERABLE status within 24 hours of the low boron determination or be in at least HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

2) The volume weighted average boron concentration of the accumulators less than 1900 ppm but greater than 1800 ppm, restore the inoperable accumulator to OPERABLE status or return the volume weighted average boron concentration of the three limiting accumulators to greater than 1900 ppm and enter ACTION c.1 within 6 hours of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

*Reactor Coolant System pressure above 1000 psig.

LIMITING CONDITION FOR OPERATION (Continued)

3) The volume weighted average boron concentration of the accumulators 1800 ppm or less, return the volume weighted average boron concentration of the three limiting accumulator to greater than 1800 ppm and enter ACTION c.2 within 1 hour of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

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

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume not resulting from normal makeup by verifying the boron concentration of the accumulator solution;

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

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

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

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

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

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on Safety Injection actuation and automatic switchover to Containment Sump Recirculation test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying that each of the following pumps develops the indicated differential pressure when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump $\geq$ 2339 psid,

2) Safety Injection pump $\geq$ 1454 psid, and

3) RHR pump $\geq$ 169 psid.

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

Boron Injection Throttle Valves	Safety Injection Throttle Valves
Valve Number	Valve Number
NI-480	NI-488
NI-481	NI-489
NI-482	NI-490
NI-483	NI-491

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 335 gpm, and

b) The total pump flow rate is less than or equal to 565 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 405 gpm, and

b) The total pump flow rate is less than or equal to 660 gpm.

3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3975 gpm.

MAIN STEAM LINE ISOLATION VALVES

UNIT 1

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3. (Unit 1 only)

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.

MAIN STEAM LINE ISOLATION VALVES

UNIT 2

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3. (Unit 2 only)

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 5 seconds when tested pursuant to Specification 4.0.5.

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.

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Overtemperature ΔT

The Overtemperature Delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to response time delays associated with the RTDs mounted in thermowells, and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower Delta T trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under overpower conditions, limits the required range for overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, (2) rate of change of temperature for dynamic compensation for instrumentation delays associated with the loop temperature detectors, and (3) axial power distribution, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔTtrip provides protection to mitigate the consequences of various size steambreaks as reported in WCAP 9226, "Reactor Core Response to Excessive Secondary Steam Break."

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 WRB-1 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, and fuel fabrication parameters 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 uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes 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 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

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 Rates

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.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power, a rod drop accident of a single or multiple rods could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBR's will be greater than the design limit DNBR value.

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 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Unit 1)

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

BASES

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

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 limit of 2200°F is not exceeded.

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

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

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q^{RTP} limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued)

At power levels below APL^{ND}, the limits on AFD are defined in the COLR, i.e. that defined by the RAOC operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g. load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible; 1) RAOC, the AFD limits of which are defined in the COLR, and 2) base load operation, which is defined as the maintenance of the AFD within a COLR specified band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(z)$ less than its limiting value. To allow operation at the maximum permissible value, the base load operating procedure restricts the indicated AFD to relatively small target band and power swings (AFD target band as specified in the COLR, $APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For base load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the base load operation, a 24 hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the base load procedure. After the waiting period extended base load operation is permissible.

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: 1) outside the allowed ΔI power operating space (for RAOC operation), or 2) outside the allowed ΔI target band (for base load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) APL^{ND} (for base load operation). Penalty deviation minutes for base load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, 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 peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit. These limits are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

Each of these is 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 $\pm$ 13 steps 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.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on the figure specified in the CORE OPERATING LIMITS REPORT (COLR), RCS flow rate and 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 relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in the figure specified in the COLR, accounts for $F^N_{\Delta H}$ less than or equal to the $F^{RTP}_{\Delta H}$ limit specified in the COLR. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Margin between the safety analysis limit DNBRs and the design limit DNBRs is maintained. A fraction of this margin is utilized to accommodate the transition core DNBR penalty (2%) and the appropriate fuel rod bow DNBR penalty (WCAP - 8691, Rev. 1).

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate

BASES

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

for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of the figure specified in the COLR. Measurement errors of 1.7% for RCS total flow rate and 4% for $F^N_{\Delta H}$ have 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-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in the figure specified in the COLR. 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 measurements, 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 indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified on the figure specified in the COLR.

The hot channel factor $F^M_Q(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or base load operation, $W(z)$ or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. $W(z)$ accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by base load operation which result in less severe transient values. The $W(z)$ function for normal operation and the $W(z)_{BL}$ function for base load operation are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

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 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 is reinstated by reducing the power by 3% from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

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. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

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. 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.

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.

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.

In MODE 4, and 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 generator is less than 50°F above each of the RCS cold leg temperatures.

CORE OPERATING LIMITS REPORT

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

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference limits, target band*, and APL^{ND*} for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), W(Z)**, APL^{ND**} and $W(Z)_{BL}$** for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{L***}$, or $F_{\Delta H}^{RTP****}$, and Power Factor Multiplier, $MF_{\Delta H}$,**** limits for Specification 3/4.2.3.

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

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

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

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

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

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

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

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

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

*** Reference 1 is not applicable to $F_{\Delta H}^{L}$.

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

CORE OPERATING LIMITS REPORT

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

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

5. DPC-NE-2011P, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March, 1990 (DPC Proprietary).

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

6. DPC-NE-3001P, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," March, 1991 (DPC Proprietary).

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

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

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

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

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

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

   (Modeling used in the system thermal-hydraulic analyses)

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

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 128 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 110 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 June 26, 1991, as supplemented September 16, 1991 and November 7, 1991, the Duke Power Company (licensee or DPC) submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes are to support the McGuire Unit 1 Cycle 8 reload with B&W fuel. In addition, DPC performed the reload analysis with DPC methodology and in the process revised several input assumptions in the Chapter 15 accident analyses. The September 16, 1991 and November 7, 1991, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

This amendment is in response to a change in the fuel design and in the supporting analytical methodology for the McGuire Units. For both Unit 1 and for Unit 2, during fuel cycles 1 through 7 the fuel was supplied by Westinghouse, (with the exception of demonstration fuel assemblies) and the supporting analyses were based principally on Westinghouse methodology. Beginning with Cycle 8, fuel manufactured by the B&W Fuel Company (BWFC) will be utilized. For Unit 1's Cycle 8, which will begin in December 1991, this will result in a mixed core of BWFC and Westinghouse fuel. The analytical methodology for this cycle has been developed by the BWFC and Duke Power Company (DPC). This methodology has been reviewed and approved by the NRC staff in response to a series of BWFC and DPC topical reports as referenced in the application and in this evaluation. Unit 2 continues at this time to operate with Westinghouse fuel based on Westinghouse developed analytical methodology. DPC plans to transition Unit 2 to BWFC fuel based on BWFC and DPC analytical methodology beginning with Cycle 8 which is scheduled to begin in early 1992. Therefore, to accommodate this transition, it is necessary to reflect TS limits for each unit based on its supporting analytical methodology. This is done by including where necessary (primarily in TS Sections 2.0 Safety Limits and Limiting Safety System Settings, 3/4.2 Power Distribution Limits, 3/4.3 Instrumentation and the BASES) separate TS pages for each unit marked to show their unit applicability.

The DPC has also utilized this amendment to include changes which are equally applicable to Units 1 and 2 and therefore do not require separate TS pages for each unit. These include: (a) deletion of obsolete references in Tables 2.2-1a, 2.2-1b, 3.3-2a and 3.3-2b regarding the now removed resistance temperature detector (RTD) bypass system, (b) a nomenclature change in event titles and deletion of a non-applicable event in Table 3.1-1, (c) a change to TS 3.4.1.2 to require 3 operable reactor coolant pumps in Mode 3, (d) a change to TS 3.5.1.1 to the required average cold leg accumulator boron concentration and a change in nomenclature, (e) a change to TS 4.5.2 in the charging, safety injection and RHR pump head, and (f) a change to TS 6.9.1.9 to reflect the addition of references for the Unit 1 Core Operating Limits Report. These changes are addressed in the following safety evaluation.

Several changes proposed by DPC are not included in this amendment. These include the changes to the tolerance ranges for the pressurizer safety valves for Units 1 and 2 (TS 3.4.2.1 and 3.4.2.2) and for the main steam safety valves for Unit 1 (TS 3.7.1.1). These proposed changes will be addressed separately as the NRC staff completes its review of them.

2.0 EVALUATION

The DPC submittal contains Technical Specifications (TS) changes, changes to the Core Operating Limits Report (COLR), markups of the appropriate FSAR Chapters, and design information relative to Cycle 8 reload. The McGuire Unit 1 plant operated in Cycle 7 with 190 Westinghouse 17x17 Optimized Fuel Assemblies (OFAs), and 3 Mark-BW 17x17 demonstration assemblies. This reload is the first time that a complete batch of B&W Fuel Company's Mark-BW 17x17 fuel design will be used at McGuire. A complete reload batch is operating at Catawba 1 in its current cycle. The use of Mark-BW fuel design in Catawba and McGuire plants has been previously approved by the NRC via the topical reports BAW-10173-P-A, Revision 2 and BAW-10174-A Revision 1.

The McGuire Cycle 8 reload is the first time DPC has performed reload safety analysis for its Westinghouse Units. Instead of relying on the B&W Fuel Company for reload design and services, DPC has developed its own reload methodology to support the use of Mark-BW fuel design in Catawba and McGuire plants. The methods and analytical models used by DPC for McGuire Unit 1 Cycle 8 fuel assembly mechanical design, nuclear design, thermal-hydraulic analyses, and non-LOCA safety analysis have been approved by the NRC.

2.1 Fuel System Design

The Mark-BW 17x17 fuel assembly design is similar in design to the Westinghouse Standard 17x17 fuel assembly design. The unique features of the Mark-BW 17x17 design include the Zircaloy intermediate spacer grids, the spacer grid restraint system, and the use of Zircaloy grids with standard lattice design.

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

2.2 Nuclear Design

The core physics parameters for Cycle 8 were generated by DPC with the PDQ07 and EPRI-NODE-P computer codes using the methodology described in the approved topical report DPC-NE-2010-A. The Reactor Protection System limits and Operational limits for the core were verified through analysis of the Cycle 8 nuclear design using the methodology described in the approved topical report DPC-NE-2011-P-A.

2.3 Thermal-Hydraulic Design

The thermal-hydraulic analyses supporting Cycle 8 operation were performed by DPC with the VIPRE-01 computer code and their approved statistical core design (SCD) methodology. The statistical core design methodology is a technique that statistically combines uncertainties associated with the core statepoint parameters, code/model, and CHF correlation to determine a statistical DNBR limit (SDL). The SDL for use with the BWCMV CHF correlation in VIPRE-01 is determined to be 1.40. To provide design flexibility, a 10.7% margin is added to the SDL which yields a design DNBR limit (DDL) of 1.55 for the generic Mark-BW and McGuire Unit 1 Cycle 8 analyses. Reactor core safety limits for Cycle 8 were generated utilizing BWCMV CHF correlation and SCD methodology with a 1.55 DDL, for a full core of Mark-BW assemblies and a radial enthalpy rise hot channel factor of 1.50.

The hydraulic compatibility of the Mark-BW and the OFA assemblies had been addressed in the approved topical report BAW-10173-P-A Revision 2. The results of the hydraulic compatibility test indicated that the total pressure drop across the Mark-BW fuel is 2.4% lower than total pressure drop across the OFA fuel. The approach taken by the licensee to address the transition core penalty is similar to that used in the topical report BAW-10173-P-A, Revision 2. The licensee determined a generic transition core penalty by modelling a conservative core configuration with one OFA assembly as the hot assembly located in a Mark-BW core. Bounding power shapes during normal and accident conditions were analyzed, yielding a maximum DNBR penalty of 3.8% for OFA fuel. The licensee address the transition core penalty for OFA fuel by applying the 3.8% DNBR penalty against the 10.7% generic margin included in the design DNBR limit.

2.4 Accident Analyses

2.4.1 Non-LOCA Analysis

The effects of this reload on each Non-LOCA event previously analyzed in the FSAR have been evaluated. The following transients were reanalyzed to account for the differences in the core physics parameters of the Mark-BW fuel and the changes in TSs:

(1) Steam System Piping Failure;
(2) Feedwater System Pipe Break;
(3) Partial Loss of Forced Reactor Coolant Flow;
(4) Complete Loss of Forced Reactor Coolant Flow;

(5) Locked Rotor;
(6) Uncontrolled Bank Withdrawal from Subcritical;
(7) Uncontrolled Bank Withdrawal at Power;
(8) Dropped RCCA/RCCA Bank;
(9) Single RCCA Withdrawal;
(10) RCCA Ejection;
(11) Turbine Trip.

The methods and results for analyses of the Steam System Piping Failure, Rod Ejection, and Dropped RCCA/RCCA Bank transients are documented in DPC topical report DPC-NE-3001 and follow-up correspondence from DPC which have been reviewed and found acceptable by the NRC staff. These analyses show that all acceptance criteria for these transients continue to be met and previous conclusions in the FSAR regarding these events remain valid for Cycle 8 operation. In addition, a bounding approach has been used to perform the analyses making them new reference analyses to be used in future reload evaluations.

The analysis of transients other than the three discussed above have been performed with RETRAN-02 NSSS transient analysis models and VIPRE-01 thermal-hydraulic subchannel analysis model described in DPC topical report DPC-NE-3000. These models have been reviewed and approved for use by the staff with certain limitations. The basis for choosing the initial conditions and assumptions used in the analysis of each transient are provided in DPC topical report DPC-NE-3002 and found acceptable by the staff with certain limitations. The specific analyses of the transients for Cycle 8 reload core conditions and operating limits were presented to the staff in a meeting with the licensee on October 7-8, 1991, and documented in follow-up submittals dated October 16, 1991 and November 5, 1991. The results of these analyses have been reviewed by the staff and found to be entirely consistent with results of reference analyses performed for a similar reload core configuration by the Babcock and Wilcox Fuel Company. These references analyses have been reviewed by the staff previously and found acceptable for referencing for either McGuire or its sister plant Catawba. The results of McGuire Cycle 8 analyses performed by DPC show that acceptance criteria for Departure from Nucleate Boiling (DNB), Peak Clad Average Temperature, Fuel Centerline Temperature, Hot Leg Boiling, Peak Reactor Coolant System Pressure and Peak Secondary System Pressure for all these transients continue to be met and previous conclusions in the FSAR regarding these transients remain valid for Cycle 8 operation.

2.4.2 LOCA Analysis

The LOCA analysis for McGuire Unit 1 transition cores with mixed Mark-BW and OFA assemblies and future cores with all Mark-BW fuel has been reviewed previously by the NRC and found acceptable.

2.5 Technical Specification (TS) Changes

The major portion of the TS changes stem from changes in safety analysis methodology, especially the power distribution and operational control methodology. Additional changes were made to reflect the existing and revised safety analyses input assumptions, provide operational flexibility or reduce the potential for a spurious trip, and correct existing errors or non-conservatisms in existing TSs. All the TS changes have been reviewed with the exception of those associated with increased tolerances on the pressurizer safety valve and main steam line safety valve lift setpoints. The changes in safety valve lift setpoints will be reviewed in 1992.

2.5.1 Changes due to Revised Power Distribution and Operational Control Methodology

The primary cause for most of the significant changes to the McGuire 1 Cycle 8 technical specifications is the new core power distribution related core operating limit methodology, developed by DPC for use in McGuire and Catawba reload analysis and cycle operation control. These revisions to the TS involve both format and value changes to the Limiting Conditions of Operation and surveillance requirements. The power distribution TS (3/4.2) are the specifications primarily affected by the methodology changes with the principal changes being to TS 3/4.2.2 and 3/4.2.3 which involve F_Q and $F_{delta-H}$ limits and surveillance. The methodology changes affecting TS 3.2 (and 2.1) are presented (primarily) in the staff approved DPC topical report DPC-NE-2011-P-A with additional related information in topical reports DPC-NE-3001-P-A and DPC-NF-2010-P-A.

The operational methodology used by the DPC is similar to that used by the B&W Fuel Company (BWFC) (described in BAW-10163P-A) for analysis, control and surveillance of BWFC reloads in Westinghouse reactors, and used by BWFC for DPC for the Catawba 1 Cycle 6 reload. The use of the BWFC methodology and the resulting TS changes was reviewed by the staff and approved in Amendment No. 88 to the Catawba operating license. Since the Catawba 1 operational methodology is similar to that developed for McGuire 1, the TS changes required to implement the methodologies are also similar for both reactors. The changes proposed for the McGuire TS relating to methodology changes, affecting primarily TS 2.1 (Safety Limits) and TS 3/4.2.1 through 3/4.2.5 (Power Distribution Limits), are essentially same as the changes approved for Catawba 1. Approximately the same revisions, deletions and additions are made, and very similar justifications are presented for the changes. There are only a few differences in terminology, format, formulation and numerical values of parameters and limits as needed to accommodate the differences in cycle parameters for the two reactors and small differences in details of the methodologies between the DPC and BWFC versions of the operational methods.

The methodology adopted by DPC for McGuire 1 for power distribution operational limits control and surveillance has been approved by NRC review and is applicable to McGuire 1. The proposed TS flow directly from the approved methodology and are directly parallel to the approved TS for Catawba 1. The relationship of the McGuire 1 TS changes to the methodology has been explained by DPC in the TS justification section of the DPC submittal and the staff review has found the justifications to be acceptable. Deviations between the two sets of TS have been examined by the staff in this review and found to be reasonable and acceptable. This review has concluded that this methodology change and the expression of that change in the revised TS for McGuire 1 Cycle 8, is also acceptable.

The following changes have been proposed for the McGuire 1 TS due to changes to the power distribution/operational control methodology.

The administrative changes to distinguish the revised Power Distribution TS for Unit 1 from the TS for Unit 2 are acceptable.

<u>TS Table 2.2-1</u> - The OTDT f_1(delta-I) limits were changed and an axial imbalance penalty, f_2(delta-I), is applied to the OPDT reactor trip to produce a reactor trip on high AFDs for credible overpower events protected by the OPDT trip function. These changes to OTDT f_1(delta-I) and OPDT f_2(delta-I) are part of the approved methodology provided in DPC-NE-2011-P-A and are acceptable.

<u>TS 3/4.2.1 Axial Flux Difference</u> - The references to the current Westinghouse methodology features such as RAOC and baseload operation are removed and replaced with an envelope of allowed axial flux difference values at various powers, following the DPC methodology. The axial flux difference setpoint envelope is provided in the Core Operating Limit Report. These changes are in accord with the approved methodology and are acceptable.

<u>TS 3/4.2.2 Heat Flux Hot Channel Factor; and TS 3/4.2.3 Nuclear Enthalpy Rise Hot Channel Factor</u> - The Westinghouse methodology is removed from both of these TS and the DPC methodology is inserted. These two sets of changes constitute the primary changes to allow adoption of the new methodology. The changes are similar to those approved for Catawba 1. The details and basis of the changes were described in the justification section of the DPC McGuire 1 submittal and the approved methodology on which the changes were based is provided in DPC-NE-2011-P-A. The review has indicated that the TS changes have been appropriately described and justified and are in accord with the methodology and are therefore acceptable.

<u>TS 3/4.2.4 Quadrant Power Tilt</u> - The TS was changed to indicate that the required reduction in thermal power below rated power for a quadrant power tilt begins, in the DPC methodology, when tilt exceeds 102 percent, rather than when exceeding 100 percent as with the Westinghouse methodology. This is in accord with the approved methodology and is acceptable.

TS 3/4.2.5 DNB Parameters - The reactor coolant flow rate limit is moved from TS 3.2.3 to this TS, and is incorporated in a new Figure 3.2-1, where it is combined with the power level to provide permitted, restricted or prohibited operating regions. This figure defines trade-offs in power and flow and has been verified by a number of thermal evaluations. It provides comparable margins to those provided in the previous Westinghouse design TS 3.2.3, which is now revised. The change is in accord with the approved methodology and is acceptable.

The Bases for the safety limits and power distribution TS which have been changed for McGuire 1 have also been revised to reflect the new methodology. These revisions present the changes and reasons for the changes in a satisfactory manner and are acceptable.

2.5.2 COLR TS Change

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The approved cycle-specific core operating limits in Amendment No. 105 (Unit 1), were revised to reflect the use of the new approved reload analysis methodology as follows:

(a) Specification 3/4.2.1

The axial flux difference limit for this specification and for this surveillance requirement is specified in the COLR. Due to the use of the new approved reload analysis methodology the target band and the base load operation are no longer applicable and are deleted.

(b) Specification 3.2.2 and Surveillance Requirement 4.2.2

The heat flux hot channel factor (F_Q) limit at rated thermal power, the normalized F_Q limit as a function of core height K(z) for both Mark-BW and OFA fuel, $F_Q$xyz limits for the operational and RPS design peaking and adjustment to K_1 value from OT delta-T KSLOPE for this specification and for this surveillance requirement are specified in the COLR.

(c) Specification 3.2.3 and Surveillance Requirement 4.2.3

The nuclear enthalpy rise hot channel factor limit ($F^L_{\Delta H}$ = MARP) for this specification (LCO) and for this surveillance (SURV), thermal power reduction (RRH) and reduction in OT delta-T K_1 set point (TRH) are specified in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

(2) Specification 6.9.1.9 Core Operating Limits Report of the Administrative Controls section of the TS for Units 1 and 2 is revised to include currently proposed TS changes and to add additional NRC approved methodologies to support the values of cycle-specific parameter limits that are applicable for the current fuel cycle. The approved methodologies are the following:

(a) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

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

(b) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," June 1983 (W Proprietary).

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

(c) WCAP-10266-P-A Rev. 2, "The 1981 Version of Westinghouse Evaluation Model Using BASH Code," March 1987, (W Proprietary).

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

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

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

(e) DPC-NE-2011P, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March 1990 (DPC Proprietary).

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

(f) DPC-NE-3001P, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," March 1991 (DPC Proprietary).

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

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

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

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

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

(i) DPC-NE-3000, Rev. 1, "Thermal-Hydraulic Analysis Methodology," May 1989.

(Modeling used in the system thermal-hydraulic analyses.)

This specification continues to require that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC.

Based on our review, the NRC staff concludes that the modifications proposed by the licensee are in accordance with the the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change does not have an adverse impact on plant safety. Accordingly, the staff finds that the proposed changes are acceptable.

2.5.3 Other TS Changes

(1) Core Safety Limits (Figure 2.1-1)

Figure 2.1-1 of the TS is revised to reflect the use of BWCMV CHF correlation and DPC's statistical core design methodology with a 1.55 design DNBR limit. The revised core safety limits are based on a full core of Mark-BW assemblies. The licensee addresses the transition core penalty for OFA fuel by applying a 3.8% DNBR penalty against the 10.7% generic margin included in the design DNBR limit. Since the BWCMV CHF correlation and the SCD methodology are approved we conclude that the revised core safety limits are acceptable for Unit 1; these changes are not applicable at this time to Unit 2.

(2) Changes to TS Table 2.2-1

The overtemperature delta-T (OTDT) and overpower delta-T (OPDT) trip function K values on TS Table 2.2-1 are revised to reflect the use of the approved BWCMV CHF correlation and the approved statistical core design methodology with a 1.55 thermal design DNBR limit. The revised overtemperature and overpower trip function K values are used in the revised analysis performed by DPC, this change is acceptable.

The administrative change for both Units 1 and 2 to delete the reference to the RTD Bypass System reflects the removal of that systems. This change is acceptable.

(3) Changes to TS Table 3.1-1 for Units 1 and 2

TS Table 3.1-1 is revised to include all accident analyses that would require reevaluation in the event that one full-length Rod Cluster Control Assembly is inoperable. Deletion of large break LOCA analysis from TS Table 3.1-1 is acceptable since LBLOCA analysis does not take credit for any control rod insertion. This change is acceptable.

(4) Changes to Delete Power Range Neutron Flux Negative Rate Trip

The changes to delete the Power Range Neutron Flux Negative Rate trip function from TS Tables 2.2-1, 3.3-1, 3.3-2, and 4.3-1 is acceptable for Unit 1 since no credit is taken for this trip function in control rod drop accidents or any other FSAR licensing basis accidents; this change is not applicable at present to Unit 2.

(5) Changes to TS Table 3.3-4

The low steam line pressure setpoint for safety injection and main steam line isolation is revised from 585 psig to 775 psig, the allowable value is revised to 755 psig. The dynamic compensation of steam pressure signal is eliminated. The core cooling analysis for the steam line break event was reanalyzed assuming an uncompensated low steam line pressure setpoint of 700 psig. The reanalysis indicated that DNB does not occur for this Condition IV event. Therefore, the setpoint change is acceptable with respect to core protection.

(6) Changes to TS Table 3.3-5

The response time for feedwater isolation is revised from 9 to 12 seconds, and the response time for steam line isolation is revised from 7 to 10 seconds. The extended response times are assumed in the core cooling reanalysis for the steam line break event for the revised low steam line pressure setpoint. The reanalysis indicated that DNB does not occur for this Condition IV event. Therefore, the response time change is acceptable with respect to core protection.

(7) Changes to TS 3.4.1.2 for Units 1 and 2

The number of operable loops required in Mode 3 operation is revised from two to three to reflect the revised analysis for the uncontrolled bank withdrawal from subcritical or low power startup condition event. This change is acceptable.

(8) Changes to TS 3.5.1.1 for Units 1 and 2

The required average accumulator boron concentration in ACTIONS c.2 and c.3 is revised from 1500 to 1800 ppm, and the basis for averaging is revised to all four accumulators instead of the limiting three.

An increased average accumulator boron concentration will ensure long term subcriticality following a LOCA. This change is acceptable.

Changing the basis for the volume weighted average to four accumulators instead of the limiting three is acceptable since all the accumulators will be emptied into the containment sump in the long term following a LOCA.

(9) Changes to TS 4.7.1.4

The permissible stroke time for main steam line isolation valves to close is revised from 5 to 8 seconds. The longer isolation time has been assumed in the main steam line break core cooling analysis. The results of the analysis show the consequences of the accident with respect to core protection to be acceptable; and therefore, the longer stroke time is considered acceptable.

(10) Changes to TS 4.5.2 (f & h) for Units 1 and 2

The ECCS pump performance requirements are revised. The required developed head and delivered flow specifications for centrifugal charging pump, safety injection pump, and residual heat removal pump were revised to provide test margin. Since the ECCS pump performance at the revised TSs values meets all acceptance criteria in both the current FSAR analyses, and in the revised analysis performed by DPC, this change is acceptable.

2.5.4 Containment Response for TS 4.7.14, TS Table 3.3-4, TS Table 3.3-5

The proposed TS changes to increase the MSIV stroke time from 5 to 8 seconds and to increase the response time for steam line isolation from 7 to 10 seconds would delay the completion of MSIV isolation following a MSLB, and therefore, could extend the period of blowdown. The additional steam released into containment would increase the peak containment temperature. The staff concern is whether the equipment needed for accident mitigation is qualified to the elevated containment temperature. To resolve the concern, the licensee and the staff held several telecons which are summarized in the licensee's letter of November 5, 1991.

The licensee indicated that while the delay of MSIV isolation would increase the blowdown mass, it would also reduce the primary coolant temperature, thereby reducing the enthalpy of the steam in the faulted steam generator. This would in turn, reduce the specific energy of the steam released to the containment. The licensee estimated that the net effect of the additional 3-second blowdown from intact steam generators and the decrease in the specific energy of the blowdown from the faulted steam generator would be to slightly increase the peak containment temperature by less than 1°F for the limiting design basis MSLB of 0.6 ft^2 break. Based on the peak containment temperature of 326°F (shown in Figure 6.2.1-16 of the McGuire FSAR) and the equipment qualification temperature of 340°F (stated in the response in November 5, 1991, letter), there is sufficient margin to accommodate the slight increase of containment temperature. Therefore, the staff concludes the TS changes to increase MSIV stroke time and steam line isolation response time are acceptable.

Based on the licensee's assessment, the proposed TS changes to remove the lead/lag dynamic compensation on the steam line pressure signal and to increase the low steam line pressure setpoint for main steam isolation from 585 psig to 775 psig would result in later main steam isolation for large steam line breaks and earlier isolation for small breaks. Therefore, for small steam line breaks, there is no adverse impact to containment temperature.

The steam line isolation is actuated by low steam line pressure or high-high containment pressure. For large breaks, the removal of the lead/lag compensation results in a delayed isolation on low steam line pressure even with the increased setpoint of 775 psig. However, for the breaks analyzed in the FSAR, the main steam isolation occurs on high-high containment pressure at essentially the same time as isolation on lead/lag compensated steam line pressure. Therefore, the mass and energy release data reported in the FSAR are still valid, and the containment temperature is not affected.

Feedwater isolation in a MSLB is actuated by, among other signals, high containment pressure and low steam line pressure. The combined effect of longer feedwater isolation response time (from 9 to 12 seconds) and removal of lead/lag compensation of the steam line pressure signal is to delay feedwater isolation following a MSLB. However, this delay does not have an adverse impact on the steam blowdown because isolation on high containment pressure occurs before the time assumed in the FSAR for isolation on low steam line pressure. The licensee's assessment for a spectrum of breaks indicated that for all, except the 1.4 ft^2 large breaks, feedwater isolation is completed in a shorter period of time. The 1.4 ft^2 break is not a limiting break. For the limiting break of 0.6 ft^2, the high containment pressure signal occurs at 2.5 seconds and the isolation is completed in 14.5 seconds. In the current FSAR case, the low steam line pressure signal occurs at 8.2 seconds and the isolation is completed in 15.2 seconds. Therefore, there is no adverse impact on the mass and energy releases resulting from the longer feedwater isolation time and the removal of the dynamic compensation of steam line pressure.

Based on the above evaluation, the staff concurs with the licensee that the above proposed TS changes have insignificant adverse impact on peak containment temperature, and the equipment qualification temperature of 340°F adequately bounds the containment temperature profile. Therefore, the staff concludes that the above proposed TS changes are acceptable.

2.5.5 Radiological Consequences

The NRC staff reviewed the FSAR markups provided as part of the M1C8 TS submittal to ascertain whether the M1C8 changes affect our assumptions and parameters used to assess the radiological consequences of the Chapter 15 accident analyses. We find that our assumptions and conclusions regarding radiological consequences stated in NUREG-0422 dated March 1978 and Supplement 4 dated January 1981 remain unaltered. Furthermore, the FSAR markups are consistent with the applicable assumptions and parameters used to support Amendment Nos. 122/104 and 118/100 to the TS for McGuire Nuclear Station, Units 1 and 2, respectively. The radiological consequences from the postulated Chapter 15 accident analyses continue to meet the regulatory criteria previously applied to the McGuire station and are, therefore, acceptable.

2.5.6 Conclusion

We have reviewed the licensee's submittal in support of the McGuire Unit 1 Cycle 8 operation with B&W Fuel Company's Mark-BW fuel and other changes applicable to both Units 1 and 2 as noted above. We have concluded that McGuire Unit 1 Cycle 8 operation, Mark-BW fuel assembly mechanical design, nuclear design, thermal-hydraulic analyses and accident analysis for McGuire Unit 1 Cycle 8 are acceptable. Technical specification changes proposed in the submittal including the other dual unit changes have been fully approved with the exception of the proposed increase in the tolerances for the pressurizer safety valve and main steam line safety valve lift setpoints. The review of these proposed changes will be completed in 1992.

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 (56 FR 47233). 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.

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