ATTACHMENT 1

Proposed McGuire Unit 1 and 2 Technical Specification Changes

NO CHANGES
FOR INFO ONLY

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

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

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATIO 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 3
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 design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 97,220 gpm per loop.

NO CHANGE FOR INFO ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. 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.
14. Undervoltage-Reactor Coolant Pumps	$\geq$ 5082 volts-each bus	$\geq$ 5016 volts-each bus
15. Underfrequency-Reactor Coolant Pumps	$\geq$ 56.4 Hz - each bus	$\geq$ 55.9 Hz - each bus
16. Turbine Trip		
a. Low Trip System Pressure	$\geq$ 45 psig	$\geq$ 42 psig
b. Turbine Stop Valve Closure	$\geq$ 1% open	$\geq$ 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	$\geq 1 \times 10^{-10}$ 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

NO CHANGES FOR INFO ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
c. Power Range Neutron Flux, P-8, Low Reactor Coolant Loop Flow, and Reactor Coolant Pump Breaker Position	$\leq$ 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.

NO CHANGES FOR INFO ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

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

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation,

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = 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, $\tau_3 \leq$ 6 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 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \leq$ 6 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 +9.0%; $f_1(\Delta I) = 0$, 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 +9.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.50% of its value at RATED THERMAL POWER.

NO CHANGES FOR INFO ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: OVERPOWER ΔT

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

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_0 = As defined in Note 1,

K_4 $\le$ 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}, $\tau_7 \ge 5$ 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 $T \le T''$.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

T	=	As defined in Note 1,
T"	=	$\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

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

NO CHANGES FOR INFO ONLY

NO CHANGES
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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 shown on Figure 3.2-3 for four loop operation:

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\,[1.0 + 0.3\,(1.0 - P)]}$,

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 Figure 3.2-3 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 $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

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.

LIMITING CONDITION FOR OPERATION

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 shown on Figure 3.2-3 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 of Figure 3.2.3:

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 of Figure 3.2-3 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.~~ THE BEGINNING OF EACH CYCLE.

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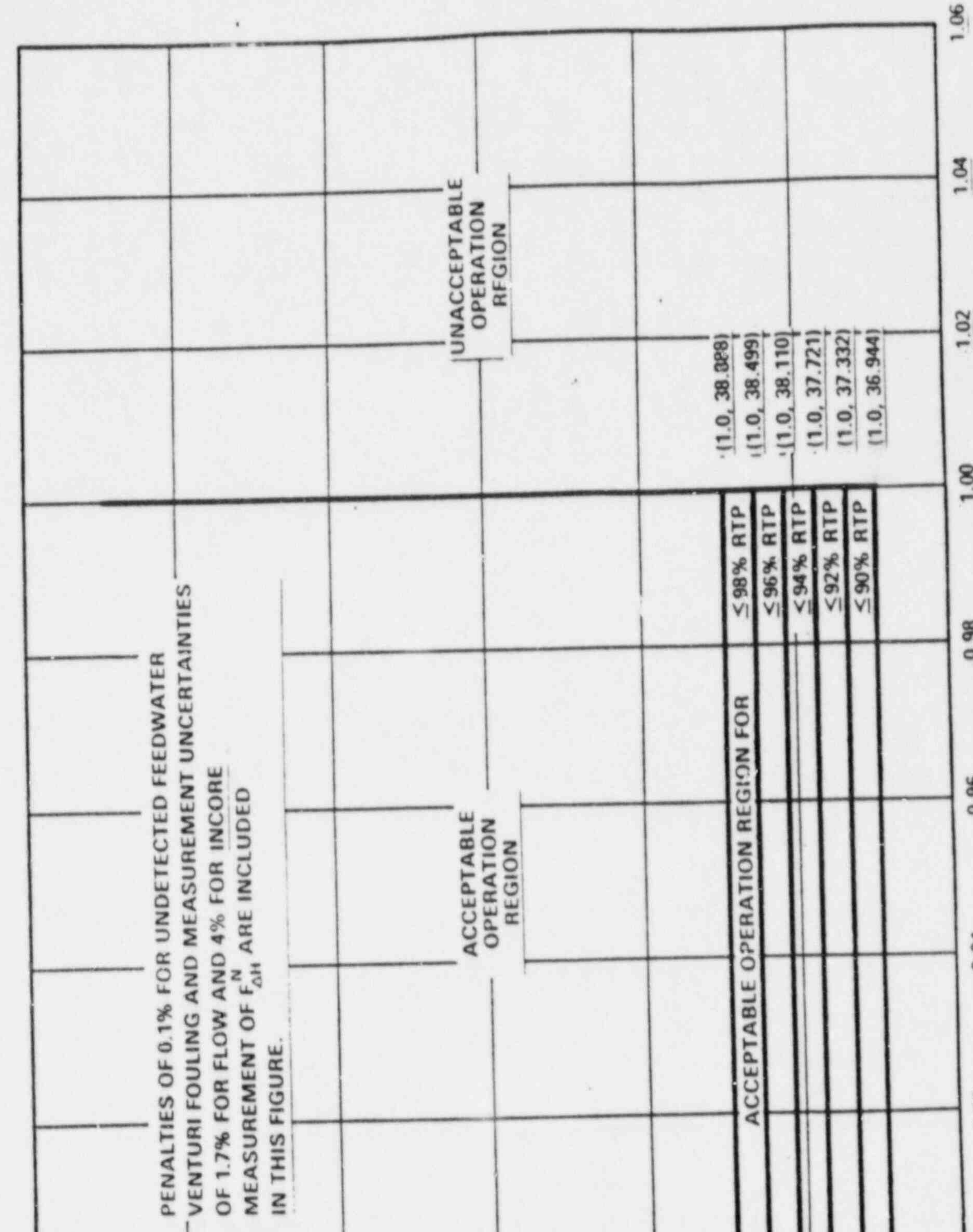

$R = F^N_{\Delta H} / 1.49[1.0 + 0.3(1.0-P)]$

Figure 3.2-3 RCS FLOW RATE VERSUS R - FOUR LOOPS IN OPERATION

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 2	1 1	2 2	1, 2 3*, 4*, 5*	1 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	(**)	(**)	(**)	(**)	(**)

NO CHANGES FOR INFO ONLY

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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

NO CHANGES
FOR INFO ONLY

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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

NO CHANGES FOR INFO ONLY

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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

NO CHANGE
FOR INFO ONLY

NO CHANGES
FOR INFO ONLY

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

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

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

#The provisions of Specification 3.0.4 are not applicable.

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

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

ACTION STATEMENTS

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

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

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

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

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

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

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

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

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

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

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

ACTION 7- Deleted

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

NO CHANGES
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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

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

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

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

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	$\leq$ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	$\leq$ 8.0 seconds*
8. Overpower ΔT	$\leq$ 8.0 seconds*
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	N.A.

*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.

NO CHANGE FOR INFO ONLY

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

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

NO CHANGES FOR INFO ONLY

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R (11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	M	N.A.	N.A.	1, 2
Low Setpoint	S	R(4)	M	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
4. 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(15)	M	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R(15)	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..	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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

NO CHANGES FOR INFO ONLY

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TABLE 4.3-1 (Continued)

TABLE NOTATION

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

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

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

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

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

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

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

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

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

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

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

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

(10) - Setpoint verification is not required.

TABLE 4.3-1 (Continued)

TABLE NOTATION

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

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

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

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

(15) - OVERTEMPERATURE SETPOINT, OVERPOWER SETPOINT, AND T_{avg} CHANNELS REQUIRE AN 18 MONTH CHANNEL CALIBRATION. CALIBRATION OF THE ΔT CHANNELS IS REQUIRED AT THE BEGINNING OF EACH CYCLE UPON COMPLETION OF THE PRECISION HEAT BALANCE OF SURVEILLANCE 4.2.3.5. RCS LOOP ΔT VALUES SHALL BE DETERMINED BY PRECISION HEAT BALANCE MEASUREMENT AT THE BEGINNING OF EACH CYCLE IN CONJUNCTION WITH SURVEILLANCE 4.2.3.5

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2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore, providing Trip System functional diversity. The functional capability at the specified trip settings is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the accident analysis to enhance the overall reliability of the Reactor Trip System.

The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance for all trips including those trips assumed in the safety analyses.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

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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.

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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 piping transit delays from the core to the temperature detectors (about 4 seconds), 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 all possible 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 piping delays from the core to the loop temperature detectors, and (3) axial power distribution, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP 9226, "Reactor Core Response to Excessive Secondary Steam Break."

NO CHANGES
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Pressurizer Pressure

In each of the pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full equivalent); and on increasing power, automatically reinstated by P-7.

Low Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 89% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 89% of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on loss of flow in more than one loop and below P-7 the trip function is automatically blocked.

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Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.5 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.6 second. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Turbine trip is automatically blocked by P-8 (a power level of approximately 48% of RATED THERMAL POWER with a turbine impulse chamber at approximately 48% of full power equivalent); and on increasing power, reinstated automatically by P-8.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System Instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF Instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

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Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range Reactor trip and de-energizing of the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump bus undervoltage and underfrequency, Turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power the above listed trips are automatically blocked.

P-8 On increasing power P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power the P-8 automatically blocks the above listed trips.

P-10 On increasing power P-10 allows the manual block of the Intermediate Range Reactor trip and the Flow Setpoint Power Range Reactor trip; and automatically blocks the Source Range Reactor trip and de-energizes the Source Range high voltage power. On decreasing power the Intermediate Range Reactor trip and the Low Setpoint Power Range Reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

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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.

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 ± 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 Figure 3.2-3, 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 Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. 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 (1.47 and 1.49 for thimble and typical cells, respectively) and the design limit DNBRs (1.32 and 1.34 for thimble and typical cells, respectively) 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

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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 Figure 3.2-3. 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 Figure 3.2-3. 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 shown on Figure 3.2-3.

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 is provided in the Peaking Factor Limit Report per Specification 6.9.1.9.

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3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip and Engineered Safety Features Actuation System instrumentation and interlocks ensure that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features Instrumentation and (3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. (Implementation of quarterly testing of RTS is being postponed until after approval of a similar testing interval for ESFAS.) The NRC Safety Evaluation Report for WCAP-10271 was provided in a letter dated February 21, 1985 from C. O. Thomas (NRC) to J. J. Sheppard (WOG-CP&L).

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Feature actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in-place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the

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3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, and (10) nuclear service water pumps start and automatic valves position.

Technical Specifications for the Reactor Trip Breakers and the Reactor Trip Bypass Breakers are based upon NRC Generic Letter 85-09 "Technical Specifications for Generic Letter 83-28, Item 4.3," dated May 23, 1985.

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REACTOR PROTECT[illegible] AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTAT[illegible] (Cont[illegible])

The En[illegible]eered Safety [illegible]atures Actuation System interlocks perform the following [illegible]tions:

P-4 Rea[illegible] - Actuates Turbine trip, closes main feedwater valves on T_{avg} [illegible]ow Setpoint, prevents the opening of the main feedwater valve which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 Defeats the manual block of Safety Injection actuation on low pressurizer pressure and low steamline pressure and defeats steamline isolation on negative steamline pressure rate. Defeats the manual block of the motor-driven auxiliary feedwater pumps on trip of main feedwater pumps and low-low steam generator water level.

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the steam dump system. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the steam dump system.

P-14 On increasing steam generator level, P-14 automatically trips all feedwater isolation valves and inhibits feedwater control valve modulation.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to intiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from [illegible]e o[illegible] this system accurately represent the spatial neutron flux distribution

ATTACHMENT 2

Justification and Safety Analysis

Following the startup of McGuire Unit 2/Cycle 2 in May 1985, a gradual decrease in the indicated value of the full-power Delta-T across the core was identified. The reactor core temperature difference (Delta-T) is the difference between the reactor inlet water temperature (cold leg) and the outlet temperature (hot leg). This Delta-T is used as a measure of reactor power for both the overtemperature and overpower Delta-T reactor trip setpoints (the Overtemperature Delta-T and Overpower Delta-T trip setpoints are detailed in Technical Specification Table 2.2-1), and is calibrated (scaled) to read 100% as determined by precision secondary system calorimetric measurements. In the latter part of June, approximately six weeks after power escalation commenced for Cycle 2, a precision heat balance was performed to verify Reactor Coolant System (RCS) flow (in accordance with T.S. 4.2.3.5). During this six week time frame, the indicated Delta-T had decreased linearly in each loop by approximately 1 degree-F. The precision heat balance indicated an increase in RCS flow from Cycle 1 and a corresponding decrease in the measured full-power Delta-Ts for the four loops. Because the Delta-T channels for the overtemperature Delta-T and overpower Delta-T setpoints were scaled to the full-power Delta-Ts obtained during the Cycle 1 precision heat balance, the Delta-T channels were underpredicting core power by as much as 5% rated thermal power. In view of the apparent non-conservative indication of power in the Delta-T channels, the Delta-T channels were rescaled to the more conservative, lower values of Delta-T obtained from the more recent precision heat balance. This event is more fully discussed in Licensee Event Report 370/85-24 (Attachment 2A).

Several Technical Specification problem areas surfaced as a result of the evaluation associated with the Delta-T incident at McGuire which should be revised in order to avoid future interpretation problems. Some of these Tech. Spec. problem areas and proposed resolutions are delineated below (other problem areas are being addressed by previous submittals or administratively within Duke Power):

I. Technical Specification Table 2.2-1

Notes 1 and 2 of Technical Specification Table 2.2-1 represent the overtemperature Delta-T and overpower Delta-T trip setpoints in units of degrees Fahrenheit. However, at McGuire the Delta-T channels and the overtemperature Delta-T and overpower Delta-T setpoints provide an indication in units of percent full-power Delta-T. In order to complement the manner in which the instrumentation is calibrated at the station, Notes 1 and 2 should be rewritten as follows:

Note 1: Overtemperature ΔT

$$(\Delta T/\Delta T_0)\left(\frac{1+\tau_1 s}{1+\tau_2 s}\right)\left(\frac{1}{1+\tau_3 s}\right) \leq K_{1}-\ldots$$

Note 2: Overpower ΔT

$$(\Delta T/\Delta T_0)\left(\frac{1+\tau_1 s}{1+\tau_2 s}\right)\left(\frac{1}{1+\tau_3 s}\right) \leq K_{4}-\ldots$$

Representing the setpoint equations in this manner shows that rescaling Delta-T_o (indicated Delta-T at rated thermal power) has no effect on the setpoint and simply adjusts the value of power indicated by the Delta-T channels. This proposed amendment is only a rearrangement of the setpoint equations and does not involve any technical changes to the equations themselves. In addition, the definition of Delta-T_o in the tables is relocated to maintain correspondence with the order of the term in the equations. These changes are administrative in nature and involve no safety concerns.

II. Technical Specification 4.2.3.5 and Table 4.3-1

Technical Specification Table 4.3-1 requires, on an 18 month frequency, a channel calibration of the overpower Delta-T and overtemperature Delta-T channels.

This is interpreted as requiring a verification of calibration for the circuitry that derives the Overpower Setpoint, Overtemperature Setpoint, and the Tavg channel constants as stated in Technical Specification Table 2.2.1 notes 1 and 2. This "Channel Calibration" is performed at the required 18 month frequency.

The Delta-T portion of overpower Delta-T and overtemperature Delta-T should be rescaled and recalibrated to a conservative value at the beginning of each fuel cycle prior to power escalation.

The conservative values of Delta-T are necessary based on operational experience which has shown that actual values of Delta-T at 100% may differ from one fuel cycle to the next.

The process of rescaling Delta-T to a conservative value is performed by subtracting a predetermined value, typically 1 degree-F, from the previous cycle Delta-T. This ensures that the Delta-T channels conservatively overpredict power until the cycle specific 100% values for each loop's Delta-T can be obtained. The cycle specific 100% values for each loop Delta-T's are determined during the Reactor Coolant System Flow Test which is performed as soon as possible after the unit reaches 100% power. Once the 100% Delta-T values are supplied, rescaling of Delta-T is performed, procedure changes are made incorporating these new values for use over the next fuel cycle, and calibration of the Delta-T circuit hardware is initiated and completed. The new 100% Delta-T values are also incorporated into the monthly procedures for verification of calibration under the "Analog Channel Operational Test" requirements.

Westinghouse has performed calculations which show that the preferred method to account for these differences is to rescale Delta-T to the "As-Measured" 100% value for each loop and to set the T' and the T" constants to the values as stated in Table 2.2.1 notes 1 and 2. This ensures that the assumptions in the safety analysis and operational margin are maintained. Failure to rescale Delta-T will either restrict operational margin or remove analysis margin possibly to the point where the assumptions of the analysis are violated.

The omission of the calibration of the Delta-T channels during power escalation has affected both McGuire Unit 1 and Unit 2. Although the decreasing

Delta-T in Unit 2 brought the problem into view, the Unit 1 channels had not been calibrated during the startup of each fuel cycle (however, since the full power Delta-T at Unit 1 had not changed dramatically over the first three cycles, the Delta-T channel errors were not of the magnitude present at Unit 2). The channel calibration was being performed on an 18 month basis without regard to the cycle startup requirements. Calibration of the Delta-T channels to the new 100% values should only be necessary at the beginning of each cycle, as it is expected that all drifts and fluctuations over the course of a cycle should remain within the allowances assumed in the Safety Analysis for that cycle.

Aside from rescaling the full power Delta-T, rescaling and recalibration of the Overpower Setpoint, Overtemperature Setpoint, and Tavg constants need only be performed if changes to Technical Specification Table 2.2.1 notes 1 and 2 are made. These changes would be made due to new safety analysis or thermal hydraulic analysis for a specific upcoming fuel cycle and would be submitted to the NRC for review and approval. Recalibration of the affected setpoints would be made during the unit shutdown prior to startup.

Accordingly, the proposed amendments to Technical Specification 4.2.3.5 and Table 4.3-1 reflect this position. (Duke has been administratively implementing these requirements since the McGuire incident). These items are clarification of the intent and do not involve relaxation of any existing requirements, and in fact are more conservative/restrictive since the current specifications do not specifically require rescaling at the beginning of each fuel cycle or if more than one cycle occurs within an 18 month span.

III. Technical Specification Table 3.3-1

Technical Specification Table 3.3-1 Action Statement 7 is revised to read "deleted" rather than "delete" to better reflect that a previously existing action statement had been deleted (ref. McGuire License Amendments 54(Unit 1)/35(Unit 2)) and is not an action to delete something. This change is administrative in nature and involves no safety concerns. Note: This change is not related to the previously discussed Delta-T event.

Based upon the preceding justification and safety analysis, Duke Power Company concludes that the proposed amendments are necessary and will not be inimical to the health and safety of the public. No changes to the Technical Specification Bases are necessitated as a result of the proposed amendments.

ATTACHMENT 3

Analysis of Significant Hazards Consideration

As required by 10CFR 50.91, this analysis is provided concerning whether the proposed amendments involve significant hazards considerations, as defined by 10CFR 50.92. Standards for determination that a proposed amendment involves no significant hazards considerations are if operation of the facility in accordance with the proposed amendment would not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated; or 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety.

I. Technical Specification Table 2.2-1

The proposed rearrangement of the setpoint equations of Technical Specification Table 2.2-1 (along with attendant relocation of the definition of Delta-T_o) makes it easier to visualize the fact that rescaling Delta-T_o has no effect on the setpoint side of the equation but simply adjusts the value of power indicated by the Delta-T channels. Since the proposed amendment does not involve any technical changes to the equations or hardware changes in the plant, but rather is only a reordering of the equation's terms and definitions as presented in the technical specifications, the changes are administrative in nature and involve no significant hazards considerations.

The Commission has provided examples of amendments likely to involve no significant hazards considerations (48FR14870). One example of this type is (i), "A purely administrative change to Technical Specifications : For example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature". This example can be applied to the proposed reordering.

II. Technical Specification 4.2.3.5 and Table 4.3-1

The proposed amendments rewording Technical Specification 4.2.3.5 and adding a footnote to Technical Specification Table 4.3-1 require that the calibration of Delta-T channels (for the 18 month channel calibration of the overpower Delta-T and overtemperature Delta-T reactor trip system instrumentation channels) be performed at the beginning of each fuel cycle (upon completion of the precision heat balance). This ensures that assumptions in the safety analysis and operational margin are maintained since operational experience has shown that actual values of Delta-T at 100% power may differ from one fuel cycle to the next (failure to rescale Delta-T will either restrict operational margin or remove analysis margin possibly to the point where the assumptions of the analysis are violated). These changes are clarification of the intent and do not involve relaxation of any existing requirements, and in fact are more conservative/restrictive since the current specifications do not specifically require rescaling at the beginning of each fuel cycle or if more than one cycle occurs within an 18 month span.

The proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated as the changes are clarifications of intent of the current specifications and do not constitute any actual changes to plant procedures or hardware and thus can have no effect on accident causal mechanisms or consequences. No possibility of a new or different kind of accident from any accident previously evaluated is created by the proposed changes since the changes are administrative in nature and accident causal mechanisms are not affected as discussed above. The changes do not involve a significant reduction in a margin of safety because they do not involve relaxation of any existing requirements, and in fact are more conservative/restrictive than the current specification's wording.

Another Commission provided example of actions not likely to involve a significant hazards consideration is (ii), "A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: For example, a more stringent surveillance requirement". Since the proposed wording is more specific as to when the affected surveillances are to be performed, the above cited example can be applied to these amendments.

III. Technical Specification Table 3.3-1

The proposed amendment to Technical Specification Table 3.3-1 Action Statement 7 to read "deleted" rather than "delete" is a purely administrative change to better reflect that a previously existing action statement had been deleted, not an action to delete something. As such, clearly no significant hazards consideration is involved, and the previously cited commission provided Example (i) in Section I above can be applied to this amendment.

Based on the preceding analyses, Duke Power Company concludes that the proposed amendments do not involve a significant hazards consideration.

bcc: P. M. Abraham
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Attachment 2A

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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
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November 13, 1985

Document Control Desk
U. S. Nuclear Regualtory Commission
Washington, D. C. 20555

Subject: McGuire Nuclear Station
Docket No. 50-370
LER 370/85-24

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 370/85-24 concerning a gradual decrease in the indicated full power delta-T. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker/orns
Hal B. Tucker

JBD/hrp

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator
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Mr. W. T. Orders
NRC Resident Inspector
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~~8511210521~~ 6pp

NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
McGuire Nuclear Station - Unit 2	0 5 0 0 0 3 7 0	1 OF 0 5

TITLE (4)

Gradual Decrease in Indicated Full Power Delta-T

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 0	1 4	8 5	8 5	0 2 4	0 0	1 1	1 3	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
Jerry B. Day - Licensing	7 0 4 3 7 3 - 7 0 3 3

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[X] NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

Following the startup of Unit 2 Cycle 2 in May 1985, a gradual decrease in the indicated value of the full power Delta-T was identified. The indicated Delta-T decreased linearly in each loop by approximately 1 degree-F. Delta-T is used as a measure of reactor power for both the overpower and overtemperature Delta-T reactor trip setpoints. The decrease in Delta-T has caused these dynamic reactor trip functions to be improperly scaled in a non-conservative direction.

Two potential causes for the decrease have been identified: a change in hot leg temperature streaming patterns which supply coolant samples to the temperature sensors; a reduction in thermal power, possibly caused by fouling of the feedwater flow venturi meters.

The channel errors have been analyzed to determine the impact on FSAR accident analyses. The results of this evaluation indicate that only a narrow range of steam line break accidents may have exceeded the design basis during this incident. Calibration procedures will be developed to set Delta-T values for each new fuel cycle; and the possible feedwater venturi fouling will be investigated.

Following the startup of Unit 2 Cycle 2, a gradual decrease in the indicated value of the full-power reactor coolant temperature difference was identified. During the eight weeks of full-power operation following the cycle 2 refueling, the indicated Delta-T had decreased linearly in each of the four loops by approximately 1 degree-F. Delta-T is used as a measure of reactor power for both the overtemperature and overpower Delta-T reactor trip setpoints. The decrease in Delta-T had caused these dynamic reactor trip functions to be improperly scaled in a non-conservative direction. The Delta-T channels were underpredicting full core power by as much as 5%.

In evaluation which followed the discovery, two potential causes for the Delta-T decrease were identified:

- a change in the hot leg temperature streaming patterns which supply coolant samples to the temperature sensors.
- a reduction in thermal power, possibly caused by fouling of the feedwater flow venturi meters.

The Delta-T channel errors have been analyzed to determine the impact on the Final Safety Analysis Report accidents which take credit for the overtemperature and overpower trip functions. The results of this safety evaluation indicate that only a narrow range of steam line break accidents (0.4 to 0.9 square feet breaks) may have exceeded the design basis during this decreasing Delta-T incident.

Background

Delta-T is used as a measure of reactor power and is calibrated (scaled) to read 100% as determined by the precision secondary system calorimetric measurements. The scaled value is used in the overtemperature and overpower Delta T core protection circuits.

The overtemperature Delta-T trip actuation circuit loops automatically vary with: 1) coolant average temperature, 2) pressurizer pressure, and 3) axial power distribution to protect the reactor core from departure from nucleate boiling (DNB) during certain design transiests.

The overpower Delta-T trip circuit loops automatically vary with: 1) coolant temperature, 2) rate of change of average temperature, and 3) axial power distribution to protect the fuel from excessive heat generation rates (KW/ft).

The overtemperature and overpower Delta-T trip setpoints are detailed in Technical Specification Table 2.2-1.

Description of Event

The results of the evaluation have identified the two possible causes of decreased Delta-T as: 1) changes in the hot leg streaming patterns, and 2) fouling of the feedwater flow venturi nozzles.

The hot leg streaming patterns are results of fuel assembly exit temperature difference. The high flow rates and differences in individual fuel assembly outlet temperatures cause temperature stratification within the Reactor Coolant system piping. A special device is installed in the hot leg piping which samples the reactor coolant at different areas within the piping. These samples mix and pass by the hot leg temperature sensor to provide an average hot leg temperature.

The fuel assembly exit temperatures did change during the period when the Delta-T was decreasing. This is an indication that the temperature streaming patterns in the piping also changed. The evaluation states that it is possible for a 0.5 degree-F decrease in Delta-T on Unit 2 based on the data provided. This would account for about half of the total decrease.

The venturi nozzle fouling is a result of a crud buildup on the feedwater flow sensing device. The buildup affects the flow readings obtained for the heat balance measurements. The venturi is a device which develops a differential pressure when flow is passed through it. This differential pressure can be accurately scaled to indicate flow as long as the venturi dimensions are not altered. The crud buildup inside the venturi affects the accuracy of the developed pressure across the venturi. Venturi fouling has possibly caused a Delta-T decrease of approximately 0.5 degree-F at McGuire.

The total 1.0 degree-F Delta-T drift caused by hot leg streaming and venturi nozzle fouling was within the uncertainty allowances assumed in the safety analysis. The additional errors associated with not calibrating the circuits during the cycle 2 startup caused the total Delta-T uncertainty to exceed the uncertainty allowances assumed in the safety analysis. The actual error for each loop in units of percent full power was:

Loop A	Loop B	Loop C	Loop D
2.5% F.P.	5.3% F.P	5.5% F.P	3.4% F.P

These errors have been evaluated for their impact on certain Final Safety Analysis Report accidents. The evaluation results indicate that only a narrow range of steam line break accidents (0.4 to 0.9 square feet) may have been in violation of the design basis.

The hot leg streaming patterns have been identified to the industry as early as 1968. The RTD Bypass System, which is installed at McGuire, was developed to reduce the effects of the temperature gradients in the piping. Westinghouse reports that no significant problems have arisen in the industry pertaining to the streaming patterns although some variations of 0.5 degree-F or less have been reported.

The feedwater flow venturi nozzles have been reported by several nuclear plants as indicating a fouling condition. The stations affected by this condition have taken measures to compensate for the errors and are monitoring the plant parameters.

The omission of the calibration of the Delta-T channels during power escalation has affected both Unit 1 and Unit 2. Although the decreasing Delta-T in Unit 2 brought the problem into view, the Unit 1 channels had not been calibrated during the startup of each fuel cycle. This requirement was not clearly defined to the station personnel. The channel calibration was being performed on an 18 month basis without regard to the cycle startup requirements.

CORRECTIVE ACTIONS:

Immediate: None

Subsequent:
1. The results of the reactor coolant flow test and how it affected the Delta-T circuits have been reviewed.
2. A work request to recalibrate the Delta-T circuits was initiated.
3. Westinghouse reviewed the key plant parameters to evaluate the Delta-T drift and submitted a report to Duke Power.

Planned:
1. A calibration procedure to set the Delta-T to a conservative value prior to reactor startup at the beginning of each cycle will be established. Necessary controls will be implemented to ensure the work is completed prior to reactor startup.
2. A calibration procedure to set the Delta-T values for the new fuel cycle based on final measured Delta-T values will be established. Testing will determine the actual 100% power level.
3. A program to monitor the full power Delta-T during the entire fuel cycle and to report significant temperature drifts to the Reactor Safety group for analysis will be developed. Data has already begun regarding this program.
4. Measures will be taken to identify the extent of the venturi fouling and initiate actions to compensate for the apparent loss of power. Alternate tests have been initiated to determine the extent of error.

SAFETY ANALYSIS

The evaluation stated that the approximately 1 degree-F drift associated with the Delta-T channels is within the uncertainty allowances assumed in the safety analysis. Therefore, the precision heat balance performed in June may be considered valid. In addition, since the error associated with the calculated flow is within the uncertainty allowance assumed in the safety analyses, there are no flow related Tech Spec violations associated with the Delta-T drift incident. However, when the 1 degree-F drift is added to the errors associated with not calibrating the Delta-T channels during startup, the total error associated with the Delta-T channels is greater than the uncertainty allowances assumed in the safety analyses.

The Delta-T channel errors can potentially impact the FSAR accidents which take credit for a reactor trip on the overtemperature or overpower Delta-T trip functions. There is sufficient margin included in the overtemperature Delta-T setpoint calculation to account for the channel errors. Therefore, the accident analyses which take credit for a reactor trip on overtemperature Delta-T (RCCA Withdrawal at Power, RCCA Misalignment, and Boron Dilution) remain valid. However, for the overpower trip function, there is not sufficient margin in the setpoint calculation to account for the channel errors.

It should be noted that credit is not explicitly taken for the overpower Delta-T trip function in any of the accidents analyzed in Chapter 15 of the McGuire FSAR. However, WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases", shows that the overpower Delta-T trip function may be relied upon to provide DNB protection for some steam line breaks at power.

Plant-specific analyses have been performed for a spectrum of intermediate steam line breaks at power. It was determined that, for break sizes equal to or greater than 0.9 square feet, reactor trip occurred on SI actuation on low steam line pressure. For break sizes less than 0.4 square feet, either no trip was necessary, or a trip occurred on low-low steam generator level. For breaks inside containment in the size range of interest, reactor trip occurs on SI actuation on high containment pressure. Thus, the overpower Delta-T trip function provides the reactor trip only for breaks between 0.4 square feet and 0.9 square feet outside containment.

There were no pipe break events during the period which would have affected the health and safety of the public. It is also possible that other trip functions such as the high flux trip, or trading off available analytical margin may have revealed acceptable results for the narrow range of affected steam line breaks.