TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10
19. Automatic Trip and Interlock Logic	2	1	2	1, 2	7
	2	1	2	3*, 4*, 5*	10

TABLE 3.3-1 (Continued)
TABLE NOTATION

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

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

##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 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

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.

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 4 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 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)	Q	N.A.	N.A.	1, 2
Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
4. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1###, 2
5. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),Q(9)	N.A.	N.A.	2##, 3, 4, 5
6. Overtemperature ΔT	S	R(15)	Q	N.A.	N.A.	1, 2
7. Overpower ΔT	S	R(15)	Q	N.A.	N.A.	1, 2
8. Pressurizer Pressure--Low	S	R	Q	N.A.	N.A.	1
9. Pressurizer Pressure--High	S	R	Q	N.A.	N.A.	1, 2
10. Pressurizer Water Level--High	S	R	Q	N.A.	N.A.	1
11. Low Reactor Coolant Flow	S	R	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
12. Steam Generator Water Level-- Low-Low	S	R	Q	N.A.	N.A.	1, 2
13. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
14. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
15. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
16. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	N.A	N.A.	N.A.	2##
b. Power Range Neutron Flux, P-8	N.A.	R(4)	N.A	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
c. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	N.A	N.A.	N.A.	1, 2
d. Turbine Impulse Chamber Pressure, P-13	N.A.	R	N.A.	N.A.	N.A.	1
18. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 12)	N.A.	1, 2, 3*, 4*, 5*
19. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*
20. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	M(13),R(14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

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

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

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

(1) - If not performed in previous 31 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) - Deleted.

(9) - Quarterly 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. Quarterly surveillance shall include verification of the High Flux at Shutdown Alarm Setpoint of less than or equal to five times background.

(10) - Setpoint verification is not required.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Auxiliary Feedwater (continued)					
f. Station Blackout (Note 1) Start Motor-Driven Pumps and Turbine-Driven Pump					
1) 4 kV Loss of Voltage	3/Bus	2/Bus Either Bus	2/Bus	1, 2, 3	19
2) 4 kV Degraded Voltage	3/Bus	2/Bus Either Bus	2/Bus	1, 2, 3	19
g. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	2-1/MFWP	2-1/MFWP	2-1/MFWP	1, 2#	27
8. Automatic Switchover to Recirculation RWST Level	3	2	2	1, 2, 3	15b
9. Loss of Power					
a. 4 kV Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	15a
b. 4 kV Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	15a
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22
d. Steam Generator Level, P-14	3/stm gen.	2/stm gen. in any operating stm gen.	2/stm gen. in each operating stm gen.	1, 2, 3	20

TABLE 3.3-3 (Continued)

TABLE NOTATION

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

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

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

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

ACTION STATEMENTS

ACTION 14 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 15a With the number of OPERABLE channels less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours. With more than one channel inoperable, enter Specification 3.8.1.1.

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

ACTION 16 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

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

ACTION STATEMENTS (Continued)

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

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

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

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

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

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

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Automatic Switchover to Recirculation		
RWST Level	≥ 90 inches	≥ 80 inches
9. Loss of Power		
Unit 1		
a) 4 kV Loss of Voltage	3174 ± 45 volts with a 8.5 ± 0.5 second time delay	≥ 3122 volts
b) 4 kV Degraded Voltage	≥ 3678.5 volts with ≤ 11 second with SI and ≤ 600 second without SI time delays	≥ 3661 volts
Unit 2		
a) 4 kV Loss of Voltage	3157 ± 45 volts with a 8.5 ± 0.5 second time delay	≥ 3108 volts
b) 4 kV Degraded Voltage	≥ 3703 volts with ≤ 11 second with SI and ≤ 600 second without SI time delays	≥ 3685.5 volts
10. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	≤ 1955 psig	≤ 1965 psig
b. T_{avg}, P-12	≥ 553°F	≥ 551°F
c. Reactor Trip, P-4	N.A.	N.A.
d. Steam Generator Level, P-14	See Item 5b. 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 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection, Reactor Trip, Feedwater Isolation, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure--Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection		See Item 1. above for all Safety Injection Surveillance Requirements.						
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection		See Item 1. above for all Safety Injection Surveillance Requirements.						

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure--High-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Negative Steam Line Pressure Rate-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	3
e. Steam Line Pressure--Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High (P-14)	S	R	Q	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Doghouse Water Level-High (Feedwater Isolation Only)	S	N.A	N.A	R	N.A.	N.A.	N.A.	1, 2
6. Containment Pressure Control System Start Permissive/Termination	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	Q	N.A.	N.A.	N.A	N.A.	1, 2, 3
d. Auxiliary Feedwater Suction Pressure-Low	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
f. Station Blackout	N.A.	N.A.	N.A	R	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
8. Automatic Switchover to Recirculation								
RSWT Level	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
9. Loss of Power								
a. 4 kV Loss of Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A	1, 2, 3, 4
b. 4 kV Degraded Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low, Low T_{avg}, P-12	N.A.	R	Q	N.A	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Level, P-14	See Item 5b for all surveillance requirements.							

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. The NRC Safety Evaluation Reports for the WCAP-10271 series were provided in letters dated February 21, 1985 from C. O Thomas (NRC) to J. J. Sheppard (WOG), February 22, 1989 from C. E. Rossi (NRC) to R. A. Newton (WOG), and April 30, 1990 from C. E. Rossi (NRC) to G. T. Goering (WOG).

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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DUKE POWER

Date: January 13, 1995

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

Subject: McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 50-370
Proposed Changes to Technical Specifications 3/4.3.1 and 3/4.3.2
Increased Surveillance Test Intervals and Allowed Outage Times for Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) Instrumentation

Gentlemen:

Please find attached an application for amendment to Facility Operating License NPF-9 and NPF-17 for McGuire Units 1 and 2, respectively. This proposed change involves increasing the surveillance test intervals and allowed outage times for RTS and ESFAS equipment based upon analyses performed by Westinghouse for the Westinghouse Owners Group (WOG) and approved by the NRC.

In February 1985, the NRC issued the Safety Evaluation Report (SER) for WCAP-10271 and WCAP-10271, Supplement 1. The SER approved quarterly testing, six hours to place a failed channel in the tripped mode, increased allowed outage times for test and maintenance, and testing in bypass for analog channels of the RTS. By letters dated April 7, 1986 and April 22, 1986, the NRC issued Amendment Nos. 54 and 35 to the facility operating licenses for McGuire Units 1 and 2 , respectively, which implemented most of the items approved in the SER for the RTS.

In February 1989, the NRC issued the SER for WCAP-10271, Supplement 2 and Supplement 2, Revision 1 and supplemented the SER in their letter dated April 30, 1990. The SER and SSER approved relaxations similar to the RTS for the ESFAS. The three SERs together approved all relaxations requested by the WOG in the WCAP-10271 program except the extensions requested for the reactor trip breakers and the main feedwater isolation on low T-cold coincident with high feedwater flow (this functional unit does not apply to McGuire). The NRC review of the WCAP-10271 program is therefore considered complete, as the final WCAP in the series was issued in June 1990.

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The proposed changes to the RTS and ESFAS instrumentation are based upon WCAP-10271, its supplements, and the NRC SERs described above.

The proposed Technical Specification amendments for McGuire are presented in Attachment 1. Additional details pertaining to the description of the amendment request and background are provided in Attachment 2 of this submittal. Attachment 3 contains the justification and safety evaluation and includes Duke Power Company's responses to the conditions imposed by the NRC in the SERs. Attachment 4 provides a discussion on the basis for a no significant hazards consideration determination. A copy of this amendment request is being provided to the appropriate North Carolina state official.

If you have any questions regarding this amendment request, please call Dwin Caldwell at (704) 875-4328.

Very truly yours,

T C McMeekin

T. C. McMeekin

DEC/s

Attachments

T. C. McMeekin, being duly sworn, states that he is Vice President, McGuire Nuclear Generation Department, Duke Powei Company: that he is authorized on the part of the said company to sign and file with the Nuclear Regulatory Commission this revision to the McGuire Ncclear Station Final Safety Analysil reprot and that al statements and matter set forth there in are true and correct to the best of his knowledge.

T C McMeekin

T. C. McMeekin, Vice President
McGuire Nuclear Station

Subscribed and sworn to me this 16th day of JAN., 1995

Kathy S. Maroleda
Notary Public

My commission expires:

12/13/98

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION AMENDMENTS FOR MCGUIRE

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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.

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

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

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

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10
19. Automatic Trip and Interlock Logic	2	1	2	1, 2	~~9~~ 7
	2	1	2	3*, 4*, 5*	10

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

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

~~***~~

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

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

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

ACTION STATEMENTS

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

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

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

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

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIC 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~~ INSERT A

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.

INSERT A for Page 3/4 3-7:

With the number of OPERABLE Channels one less that the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

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 4 ~~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 (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.

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TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

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

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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~~ Q	N.A.	N.A.	1, 2
Low Setpoint	S	R(4)	~~M~~ S/U (1)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	~~M~~ Q	N.A.	N.A.	1, 2
4. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1), ~~M~~ ~~Q~~	N.A.	N.A.	1###, 2
5. Source Range, Neutron Flux	S	R(4, 5)	S/U(1), ~~M(9)~~ Q(9)	N.A.	N.A.	2##, 3, 4, 5
6. Overtemperature ΔT	S	R(15)	~~M~~ Q	N.A.	N.A.	1, 2
7. Overpower ΔT	S	R(15)	~~M~~ Q	N.A.	N.A.	1, 2
8. Pressurizer Pressure--Low	S	R	~~M~~ Q	N.A.	N.A.	1
9. Pressurizer Pressure--High	S	R	~~M~~ Q	N.A.	N.A.	1, 2
10. Pressurizer Water Level--High	S	R	~~M~~ Q	N.A.	N.A.	1
11. Low Reactor Coolant Flow	S	R	~~M~~ Q	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	MODES FOR WHICH ACTUATION LOGIC TEST	SURVEILLANCE IS REQUIRED
12. Steam Generator Water Level-- Low-Low	S	R	~~M~~ Q	N.A.	N.A.	1, 2
13. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	~~M~~ Q	N.A.	1
14. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	~~M~~ Q	N.A.	1
15. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
16. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	~~M~~ N.A.	N.A.	N.A.	2##
~~b. Low Power Reactor Trips Block, P-7~~	~~N.A.~~	~~R(4)~~	~~M (8)~~	~~N.A.~~	~~N.A.~~	~~1~~
~~c.~~ b. Power Range Neutron Flux, P-8	N.A.	R(4)	~~M (8)~~ N.A.	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)

TABLE NOTATION

*	-	With the Reactor Trip System breakers closed and the Control Rod Drive System capable of rod withdrawal.
##	-	Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
###	-	Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.
(1)	-	If not performed in previous ~~7~~ 31 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)	-	Deleted. ~~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~~ Quarterly 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~~ Quarterly surveillance shall include verification of the High Flux at Shutdown Alarm Setpoint of less than or equal to five times background.
(10)	-	Setpoint verification is not required.

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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. RCS loop ΔT values shall be determined by precision heat balance measurements at the beginning of each cycle.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection, Reactor Trip, Feedwater Isolation, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High	3	2	2	1, 2, 3	15
d. Pressurizer Pressure - Low-Low	4	2	3	1, 2, 3#	19
e. Steam Line Pressure-Low					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15
Three Loops Operating	(**)	(**)	(**)	(**)	(**)

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Auxiliary Feedwater (continued)					
f. Station Blackout (Note 1) Start Motor-Driven Pumps and Turbine-Driven Pump	6-3/Bus	2/Bus Either Bus	2/Bus	1, 2, 3	19
g. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	2-1/MFWP	2-1/MFWP	2-1/MFWP	1, 2#	27
8. Automatic Switchover to Recirculation					
RWST Level	3	2	2	1, 2, 3	15b
9. Loss of Power					
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	15a
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22
d. Steam Generator Level, P-14	3/stm gen.	2/stm gen. in any operating stm gen.	2/stm gen. in each operating stm gen.	1, 2, 3	20

TABLE 3.3-3 (Continued)

TABLE NOTATION

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

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

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

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

ACTION STATEMENTS

ACTION 14 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within ~~6~~ 12 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to ~~2~~ 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within ~~1 hour.~~ 6 hours

ACTION 15a With the number of OPERABLE channels less than the total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is ~~bias~~ placed in the tripped condition within ~~1 hour.~~ 6 hours With more than one channel inoperable, enter Specification 3.8.1.1.

INSERT B

ACTION 16 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to ~~2~~ 4 hours for surveillance testing per Specification 4.3.2.1.

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

INSERT B for Page 3/4 3-23:

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

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

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

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

a. The inoperable channel is placed in the tripped condition within ~~1 hour~~ 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to ~~2~~ 4 hours for surveillance testing of other channels per Specification 4.3.1.1 and Specification 4.3.2.1.

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

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to ~~2~~ 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

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

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

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

TABLE 3.3-4 (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~~ 5b. 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 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection, Reactor Trip, Feedwater Isolation, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low-Low	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure--Low	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-High	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure--High-High	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Negative Steam Line Pressure Rate-High	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	3
e. Steam Line Pressure--Low	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High (P-14)	S	R	~~M~~ Q	N.A.	M(1) ~~N.A.~~	M(1) ~~N.A.~~	Q ~~N.A.~~	1, 2,3
c. Doghouse Water Level-High (Feedwater Isolation Only)	S	N.A	N.A	R	N.A.	N.A.	N.A.	1, 2
6. Containment Pressure Control System								
Start Permissive/ Termination	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	~~M~~ Q	N.A.	N.A.	N.A	N.A.	1, 2, 3
d. Auxiliary Feedwater Suction Pressure-Low	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
f. Station Blackout	N.A.	N.A.	N.A	R	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
8. Automatic Switchover to Recirculation								
RSWT Level	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
9. Loss of Power								
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Level, P-14	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~M(1)~~	~~M(1)~~	~~Q~~	~~1, 2, 3~~

See item 5b for all surveillance requirements.

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. INSERT C ~~(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

INSERT C for Page B 3/4 3-1:

The NRC Safety Evaluation Reports for the WCAP-10271 series were provided in letters dated February 21, 1985 from C. O. Thomas (NRC) to J. J. Sheppard (WOG), February 22, 1989 from C. E. Rossi (NRC) to R. A. Newton (WOG), and April 30, 1990 from C. E. Rossi (NRC) to G. T. Goering (WOG).

NO CHANGES THIS PAGE.
FOR INFORMATION ONLY

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.

ATTACHMENT 2

BACKGROUND AND DESCRIPTION OF AMENDMENT REQUEST

Background

The purpose of this Technical Specification amendment request is to obtain relaxation regarding the conduct of surveillance testing of the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS). As a result of concern of the impact of existing testing and maintenance requirements on plant operation, particularly in the area of instrumentation, the Westinghouse Owners Group (WOG) initiated a program to develop justification to be utilized in revising individual plant Technical Specifications. Operating plants have experienced many inadvertent reactor trips and safeguards actuations during performance of instrumentation surveillance, causing unnecessary transients and challenges to plant safety systems. Significant time and effort on the part of the plant staff was devoted to performing, reviewing, documenting, and tracking various surveillance activities, which in many instances appeared unwarranted based on the high reliability of the equipment. Significant benefits for operating plants appeared to be achievable through revision of instrumentation test and maintenance requirements. A complete chronology of the WOG efforts and interactions with the NRC is contained in a document titled "Westinghouse Owners Group Guidelines for Preparing Submittals Requesting Revision of Reactor Protection System Technical Specifications Based on Generic Approval of WCAP-10271 and Supplements" (TOPS Guidelines - August 1990).

Description of Amendment Request

The list of Technical Specification changes included in this amendment request is as follows:

(a) Changes as described in the marked-up copy of Technical Specification 3/4.3.1 (Attachment 4). These changes include:

   (i) The surveillance test interval in Table 4.3-1 for functional unit 17, Reactor Trip System Interlocks, Analog Channel Operational Test, is changed from monthly to "R" (at least once per 18 months) for each of the interlocks. Note that for functional unit 17, Reactor Trip System Interlocks, the testing required by the channel calibration encompasses the testing required by the analog channel operational test. Hence, the ACOT surveillance frequency is being changed to "N.A.", since this requirement will be covered by the channel calibration.

   (ii) Increase in surveillance intervals for Reactor Trip System (RTS) analog channel operational tests from once per month to once per quarter.

   (iii) In Table 3.3-1, new ACTION 7 is added to allow 6 hours to restore an inoperable channel to operable status before requiring shutdown to HOT STANDBY within the next 6 hours and to allow bypass of a channel for up to 4 hours for surveillance testing, provided the other channel is OPERABLE. Make new ACTION 7 applicable to functional units 16 (Safety Injection Input from ESF) and 19 (Automatic Trip and Interlock Logic), rather than ACTION 9.

(iv) In Table 3.3-1, ACTION 9 is modified to change the 2-hour allowance for bypassing one channel for surveillance testing to 4 hours. This is necessary because new ACTION 7 allows 4 hours for the SSPS. Testing of the SSPS requires bypassing the reactor trip breakers, and allowing 4 hours for the SSPS would provide no advantage unless the 4-hour stipulation were also made to apply to the reactor trip breakers.

(b) Changes as described in the marked-up copy of Technical Specification 3/4.3.2 (Attachment 4). These changes include:

(i) Increase in surveillance intervals for Engineered Safety Features Actuation System (ESFAS) analog channel operational tests from once per month to once per quarter.

(ii) Increase in the time that an inoperable ESFAS channel may be maintained in an untripped condition from 1 hour to 6 hours.

(iii) Increase in the time that an inoperable ESFAS channel may be bypassed to allow testing of another channel in the same function from 2 hours to 4 hours.

(iv) In Table 3.3-3, revise the following ACTIONS in accordance with the Westinghouse Owners Group guidelines as follows:

- ACTION 14 is changed to allow 12 hours before placing the unit in HOT STANDBY and increases from 2 to 4 hours the time that a channel may be bypassed.

- ACTION 15 is changed to increase the time that an inoperable channel may be untripped from 1 to 6 hours.

- ACTION 15a is changed to increase the time that an inoperable channel may be untripped from 1 to 6 hours.

- ACTION 16 is changed to increase the time that an additional channel may be bypassed from 2 to 4 hours.

- ACTION 19 is changed to allow the inoperable channel to remain untripped for 6 hours and to allow the inoperable channel to be bypassed for 4 hours.

- ACTION 21 is changed to allow 6 hours to restore an inoperable channel prior to placing the unit in HOT STANDBY and increases the time that a channel may be bypassed from 2 to 4 hours.

- Create new ACTION 15b to be inserted in Table 3.3-3 following ACTION

15a. ACTION 15b is made to apply to functional unit 8 (Automatic Switchover to Recirculation), as this functional unit was not part of the program for which generic NRC relief has been granted.

(v) In Table 4.3-2, a change was made which will enhance the technical specification from a human factors standpoint. Functional units 5b and 10d both describe the steam generator high-high water level (P-14) turbine trip and feedwater isolation. The conditions delineated in 10d are the most limiting and must be followed; therefore, the current conditions in 5b are being deleted and replaced by those of 10d. The surveillance requirements of 10d will then be deleted from the table

(c) Revisions to the 3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION BASES.

ATTACHMENT 3

JUSTIFICATION AND SAFETY EVALUATION

Justification and Safety Evaluation

In WCAP-10271 and its supplements, the WOG evaluated the impact of the proposed surveillance test interval (STI) and allowed outage time (AOT) changes on core damage frequency and public risk. The NRC staff concluded in its evaluation of the WOG evaluation that an overall upper bound increase of the core damage frequency due to the proposed STI/AOT changes is less than 6 percent for Westinghouse Pressurized Water Reactor (PWR) plants. The NRC staff also concluded that actual core damage frequency increases for individual plants are expected to be substantially less than 6 percent. The NRC staff considered this core damage frequency increase to be small compared to the range of uncertainty in the core damage frequency analyses and therefore acceptable.

The NRC staff concluded in addition that a staggered test strategy need not be implemented for ESFAS analog channel testing and is no longer required for RTS analog channel testing. (Since Duke Power Company has never applied for an increased surveillance test interval for the McGuire RTS, the staggered test strategy was never implemented.) This conclusion was based on the small relative contribution of the analog channels to RTS/ESFAS unavailability, process parameter signal diversity, and normal operational testing sequencing.

The NRC determined that the requirement to routinely verify permissive status is a different consideration than the availability of trip or actuation channels which are required to change state on the occurrence of an event and for which the function availability is more dependent on the surveillance interval. The definition of the channel check includes comparison of the channel status with other channels for the same parameter. For the RTS interlocks, the change from a monthly surveillance requirement to at least once every 18 months is therefore justified.

The proposed changes are consistent with the NRC staff's letters dated February 21, 1985, February 22, 1989, and April 30, 1990, to the WOG regarding evaluation of WCAP-10271, WCAP-10271 Supplement 1, WCAP-10271 Supplement 2, and WCAP-10271 Supplement 2, Revision 1. The staff has stated that approval of these changes is contingent upon confirmation that certain conditions are met. It is the interpretation of Duke Power Company that conditions imposed in the SER for WCAP-10271 and WCAP-10271 Supplement 1 for the RTS instrumentation shall also be applied to the ESFAS where appropriate. Duke Power Company's response to these conditions is provided below:

The first condition in the RTS SER required the use of a staggered test plan for the RTS channels changed to the quarterly test frequency.

Response
The NRC did not impose this requirement for ESFAS channels and it was subsequently removed for the RTS channels. Duke Power Company never applied for an amendment to change the surveillance interval for RTS channels in the past; therefore, the staggered test plan was never utilized.

The second condition in the RTS SER required that plant procedures require a common

cause evaluation for failure in RTS channels changed to the quarterly test frequency and additional testing for plausible common cause failures.

Response
In the event of failure in an RTS channel, a Problem Investigation Process (PIP) is initiated to document the failure and assess the need for additional corrective action. This corrective action includes evaluation for common cause failure mechanisms where appropriate. Testing of additional channels is conducted when there is reason to believe a common cause failure mechanism exists. Station guidelines have been developed to document the current practices reflecting the Failure and Analysis Trending Program used to document nuclear steam supply system (NSSS) and balance of plant (BOP) failures. The program database is periodically reviewed by the responsible component expert to ascertain any trends or common cause failures. In addition, records of failures in RTS channels are input into the Nuclear Plant Reliability Data System (NPRDS) and trending of RTS failures is periodically performed utilizing this database. Finally, it should be noted that in addition to actual hardware failures, problems that may be introduced into the equipment as a result of calibration and other maintenance or testing activities also are evaluated for common cause potential.

The third condition in the RTS SER required installed hardware capability for testing in the bypass mode.

Response
McGuire currently has installed bypass capability within the 7300 Protection and Control System.

The fourth condition in the RTS SER involved channels that provide input to both the RTS and the ESFAS. As stated by NRC in the safety evaluation for WCAP-10271:

"In order to avoid confusion in plant Technical Specifications regarding such dual function channels, the staff concludes that either (1) the channels should not be changed in the RTS tables until the ESFAS review is finished or (2) cautionary notes in the RTS tables should refer to the more stringent ESFAS requirements."

Response
Now that the ESFAS SER has been issued and all of the relaxations for the RTS analog channels are applicable to the ESFAS analog channels, this condition does not apply. Cautionary notes as described above have been deleted.

The fifth condition in the RTS SER, and second in the ESFAS SER, addresses setpoint drift. Confirmation is needed to show that the instrument setpoint methodology includes sufficient adjustments to offset the drift anticipated as a result of less frequent surveillance.

Response
McGuire engineering personnel have reviewed "as found" and "as left" data for the RTS and ESFAS setpoints for a 16-month period for Unit 1 and a 14-month period for Unit 2 and

concluded that sufficient adjustments are present to offset the drift anticipated as a result of quarterly surveillance. This information is available for NRC inspection.

The first condition in the ESFAS SER required that the plant-specific applications must confirm the applicability of the generic analyses to the plant.

Response

The WCAP methodology addresses two-loop, three-loop, and four-loop plants with relay or solid state systems. The RTS and ESFAS functions for which increased surveillance intervals and allowed outage times are being requested in this amendment request are those for which NRC approval has already been granted through issuance of the SERs and supplements for the basis WCAP series. Except as already described and justified in this submittal, no additional changes are being requested in this amendment request beyond those given approval by the NRC.

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION
AND ENVIRONMENTAL IMPACT ANALYSIS

No Significant Hazards Consideration Determination

The standards used to arrive at a proposed determination that the changes described involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that 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 then a no significant hazards determination can be made.

Duke Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed RTS and ESFAS Technical Specification changes for McGuire and determined that a significant hazards consideration is not involved. In support of this conclusion, the following analysis is provided.

Criterion 1 - Operation of McGuire in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The determination that the results of the proposed changes are within all acceptable criteria was established in the SERs prepared for WCAP-10271, WCAP-10271 Supplement 1, WCAP-10271 Supplement 2, and WCAP-10271 Supplement 2, Revision 1 issued by letters dated February 21, 1985, February 22, 1989, and April 30, 1990. Implementation of the proposed changes is expected to result in an acceptable increase in total RTS yearly unavailability. This increase, which is primarily due to less frequent surveillance, results in an increase of similar magnitude in the probability of an Anticipated Transient Without Scram (ATWS) and in the probability of core melt resulting from an ATWS and also results in a small increase in core damage frequency (CDF) due to ESFAS unavailability.

Implementation of the proposed changes is expected to result in a significant reduction in the probability of core melt from inadvertent reactor trips. This is a result of a reduction in the number of inadvertent reactor trips (0.5 fewer inadvertent reactor trips per unit per year) occuring during testing of RTS instrumentation. This reduction is primarily attributable to testing in bypass and less frequent surveillance.

The reduction in core melt frequency from inadvertent reactor trips is sufficiently large to counter the increase in ATWS core melt probability resulting in an overall reduction in total core melt probability.

The values determined by the WOG and presented in the WCAP for the increase in CDF were verified by Brookhaven National Laboratory (BNL) as part of an audit and sensitivity analysis for the NRC staff. Based on the small value of the increase compared to the range of uncertainty in the CDF, the increase is considered acceptable.

Changes to surveillance test frequencies for the RTS interlocks do not represent a significant reduction in testing. The currently specified test interval for interlock channels allows the surveillance requirement to be satisfied by verifying that the permissive logic is in its required state using the permissive annunciator window. The surveillance as currently required only verifies the status of the permissive logic and does not address verification of channel setpoint or operability. The setpoint verification and channel operability are verified after a refueling shutdown. The definition of the channel check includes comparison of the channel status with other channels for the same parameter. The requirement to routinely verify permissive status is a different consideration than the availability of trip or actuation channels which are required to change state on the occurrence of an event and for which the function availability is more dependent on the surveillance interval. The change in surveillance requirement to at least once every refueling does not therefore represent a significant change in channel surveillance and does not involve a significant increase in unavailability of the RTS.

The proposed changes do not result in an increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes affects the probability of failure of the RTS but does not alter the manner in which protection is afforded nor the manner in which limiting criteria are established.

Criterion 2 - The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not result in a change in the manner in which the RTS provides plant protection. No change is being made which alters the functioning of the RTS (other than in a test mode). Rather, the likelihood or probability of the RTS functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident.

The proposed changes do not involve hardware changes except those necessary to implement testing in bypass. Some existing instrumentation is designed to be tested in bypass and current Technical Specifications allow testing in bypass. Testing in bypass is also recognized by IEEE standards. Therefore, testing in bypass has been previously approved and implementation of the proposed changes for testing in bypass does not create the possibility of a new or different kind of accident from any previously evaluated. Furthermore, since the other proposed changes do not alter the functioning of the RTS, the possibility of a new or different kind of accident from any previously evaluated has not been created.

Criterion 3 - The proposed license amendment does not involve a significant reduction in a margin of safety.

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The impact of reduced testing other than as addressed above is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act. Experience has shown that the initial uncertainty assumptions are valid for reduced testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety by:

1) Less frequent testing will result in fewer inadvertent reactor trips and actuation of Engineered Safety Features Actuation System components.

2) Higher quality repairs leading to improved equipment reliability due to longer allowable repair times.

3) Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation. This is due to less frequent distraction of the operator and shift supervisor to attend to instrumentation testing.

The foregoing analysis demonstrates that the proposed amendment to McGuire's Technical Specifications does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident, and does not involve a significant reduction in a margin of safety.

Based upon the preceding analysis, Duke Power Company concludes that the proposed amendment does not involve a significant hazards consideration.

Environmental Impact Analysis

The proposed Technical Specification amendment has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. The proposed amendment does not involve a significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor increase individual or cumulative occupational radiation exposures. Therefore, the proposed amendment meets the criteria given in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.