ATTACHMENT 2

Proposed changes to Technical Specifications of Facility Operating Licenses NPF 37, NPF 66, NPF 72 and NPF 77.

Revised pages:	B 2-8	3/4 3-15
	B 2-9	3/4 3-17
	3/4 3-2	3/4 3-19
	3/4 3-3	3/4 3-20
	3/4 3-5	3/4 3-21
	3/4 3-6	3/4 3-22
	3/4 3-9	3/4 3-34
	3/4 3-10	3/4 3-35
	3/4 3-11	3/4 3-36
	3/4 3-12	3/4 3-37
	3/4 3-12a	3/4 3-38
	3/4 3-13(Bw)	B 3/4 3-1

NOTE: The attached markups reflect the incorporation of the pending amendment requests for Generic Letter 87-09 and the revision to the BDPS requirements. The following pages are affected:

Generic Letter 87-09	BDPS
3/4 3-2	3/4 3-2
3/4 3-3	3/4 3-5
3/4 3-5	3/4 3-9
3/4 3-12	3/4 3-12
3/4 3-12a	3/4 3-12a
3/4 3-15	
3/4 3-17	
3/4 3-19	
3/4 3-21	

9208100010 920805
PDR ADOCK 05000454
P PDR

ZNLD/615/120

Turbine Trip#

A Turbine trip initiates a Reactor trip. On decreasing power the Turbine trip is automatically blocked by ~~P-7 or~~ P-8 (a power level of approximately ~~10% (P-7) or~~ 30% (P-8) of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately ~~10% (P-7) or~~ 30% (P-8) of full power equivalent); and on increasing power, reinstated automatically by ~~P-7 or~~ 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.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position trips are anticipatory trips which provide core protection against DNB. The Open/Close Position trips assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Trip System. 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 more than one reactor coolant pump breaker is opened. Below P-7 the trip function is automatically blocked.

~~#A Reactor trip on Turbine trip is enabled above P-7 (10%) until the modification is implemented which enable. Reactor trip on Turbine trip above P-8 (30%).~~

LIMITING SAFETY SYSTEM SETTINGS

BASES

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 (i.e., prevents premature block of Source Range trip), provides an automatic backup block for Source Range Neutron Flux doubling, and the manual block that de-energizes the high voltage to the Source Range detectors. On decreasing power, Source Range Level trips and Neutron Flux doubling circuits 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, more than one reactor coolant pump breaker open, 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 and Turbine trip. On decreasing power, the P-8 automatically blocks the single loop low flow trip and Turbine trip.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range Reactor trip and the Low Setpoint Power Range Reactor trip; and automatically blocks the Source Range Reactor trip and provides an automatic backup function to de-energize 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 and Source Range high voltage to the detectors is restored if power decreases below the P-6 setpoint. Provides input to P-7.

P-13 Provides input to P-7.

~~#A Reactor trip on Turbine trip is enabled above P-7 (10%) until the modification is implemented which enables Reactor trip on Turbine trip above P-8 (30%).~~

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					
a. High Setpoint	4	2	3	1, 2	2~~#~~
b. 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	5
7. Overtemperature ΔT	4	2	3	1, 2	6~~#~~
8. Overpower ΔT	4	2	3	1, 2	6~~#~~
9. Pressurizer Pressure-Low (Above P-7)	4	2	3	1	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
10. Pressurizer Pressure-High	4	2	3	1, 2	6
11. Pressurizer Water Level-High (Above P-7)	3	2	2	1	6
12. Reactor Coolant Flow-Low					
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 in 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
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 (Above ~~P-7 or~~ P-8) ~~****~~					
a. Emergency Trip Header Pressure	3/Train	2/Train	2/Train	1	6
b. Turbine Throttle Valve Closure	4	4	1	1	6

~~****A Reactor trip on Turbine trip is enabled above P-7 (10%) until the modification is implemented which enables Reactor trip on Turbine trip above P-8 (30%).~~

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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

~~**The boron dilution flux doubling signals may be blocked during reactor startup.~~

~~***These channels also provide inputs to ESFAS. The Action Statement for the channels in Table 3.3-3 is more conservative and, therefore, controlling.~~

~~#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.

@Whenever the Reactor Trip Bypass Breakers are racked in and closed for bypassing a Reactor Trip Breaker.

ACTION STATEMENTS

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

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

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

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

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

ACTION 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.

TABLE 3.3-1 (Continued)
ACTION STATEMENTS (Continued)

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 restore the inoperable channel to OPERABLE status within 48 hours or within the next hour open the reactor trip breakers, suspend all operations involving positive reactivity changes, and verify valves CV-111B, CV-8428, CV-8439, CV-8441 and CV-8435 are closed and secured in position. With no channels OPERABLE verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, and take the actions stated above within 1 hour and verify compliance 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.

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.

restore the inoperable channel to OPERABLE status within 6 hours, or

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within the next 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 ~~1~~ 6 hours.

ACTION 12 - a. With one of the diverse trip features (Undervoltage or Shunt Trip attachment) inoperable, restore it to OPERABLE

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(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5a)#	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)#	Q	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. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)#	Q	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5a)#	Q	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5b, ~~12~~)#	Q(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R(13)#	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R#	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low (Above P-7)	S	R#	Q**	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R#	Q	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High (Above P-7)	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. Reactor Coolant Flow-Low	S	R~~#~~	Q	N.A.	N.A.	1
13. Steam Generator Water Level-Low-Low	S	R~~#~~	Q~~**~~	N.A.	N.A.	1, 2
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	Q~~**~~(10)	N.A.	1
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	Q(10)	N.A.	1
16. Turbine Trip (Above ~~P-7 or~~ P-8)~~###~~						
a. Emergency Trip Header Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Throttle 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 Coolant Pump Breaker Position Trip (Above P-7)	N.A.	N.A.	N.A.	R	N.A.	1
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)~~#~~	~~Q~~ R	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)~~#~~	~~Q (8)~~ R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)~~#~~	~~Q (8)~~ R	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
19. Reactor Trip System Interlocks (Continued)						
d. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	~~Q (8)~~ R	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	~~Q (8)~~ R	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (11)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	(15), R(16)	N.A.	1, 2, 3*, 4*, 5*

The initial single point comparison of incore to excore AXIAL FLUX DIFFERENCE following a refueling outage shall be performed prior to exceeding 75% of RATED THERMAL POWER. Otherwise the

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

~~**These channels also provide inputs to ESFAS. The Operational Test Frequency for these channels in Table 4.3-2 is more conservative and, therefore, controlling.~~

~~***A Reactor trip on Turbine trip is enabled above P-7 (10%) until the modification is implemented which enables Reactor trip on Turbine trip above P-8 (30%).~~

~~#The specified 18 month interval may be extended to 32 months for Cycle 1 only.~~

##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) ~~S~~single point comparison of incore to excore AXIAL FLUX DIFFERENCE shall be performed 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.

(5a) Initial plateau curves shall be measured for each detector. Subsequent plateau curves shall be obtained, evaluated and compared to the initial curves. 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.

(5b) With the high voltage setting varied as recommended by the manufacturer, an initial discriminator bias curve shall be measured for each detector. Subsequent discriminator bias curves shall be obtained, evaluated and compared to the initial curves.

(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) Not used. ~~With power greater than or equal to the interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.~~

(9) 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. ~~Surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period.~~

For the purposes of this surveillance, monthly shall mean at least once per 31 EFPD. The 24 hour completion time provisions of Specification 4.0.3 are not applicable.

For the purposes of this surveillance, quarterly shall mean at least once per 92 EFPD.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall be performed such that each train is tested at least every 62 days on a STAGGERED TEST BASIS and following maintenance or adjustment of the Reactor Trip Breakers and shall include independent verification of the OPERABILITY of the Undervoltage and Shunt Trip Attachments of the Reactor Trip Breakers.

(12) Not used. ~~At least once per 18 months during shutdown verify that on a simulated Boron Dilution Doubling test signal CVCS valves 112D and E open and 112B and C close within 30 seconds.~~

(13) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

(14) Verify that the appropriate signals reach the Undervoltage and Shunt Trip Relays, for both the Reactor Trip and Bypass Breakers from the Manual Trip Switches. ~~Initial performance of this surveillance requirement for the Reactor Trip Bypass Breakers is to be completed prior to the startup following the third refueling outage for Unit 1 and the second refueling outage for Unit 2.~~

(15) Manual Shunt Trip prior to the Reactor Trip Bypass Breaker being racked in and closed for bypassing a Reactor Trip Breaker.

(16) Automatic Undervoltage trip. ~~Initial performance of this surveillance requirement is to be completed prior to the startup following the third refueling outage for Unit 1 and the second refueling outage for Unit 2.~~

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, Start Diesel Generators, Containment Cooling Fans, Control Room Isolation, Phase "A" Isolation, Turbine Trip, Auxiliary Feedwater, Containment Vent Isolation, and Essential 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-1	3	2	2	1, 2, 3	~~15*~~ 19#
d. Pressurizer Pressure-Low (Above P-11)	4	2	3	1, 2, 3#	19#
e. Steam Line Pressure-Low (Above P-11)	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#	~~15*~~ 19#
2. Containment Spray					
a. Manual Initiation	2 pair	1 pair	2 pair	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	4	2	3	1, 2, 3	16

2

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
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	23
2) System	2	1	2	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Containment Pressure-High-2	3	2	2	1, 2, 3	~~15*~~ 19
d. Steam Line Pressure-Low (above P-11)	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#	~~15*~~ 19
e. Steam Line Pressure - Negative Rate-High (below P-11)	3/steam line	2/steam line any steam line	2/steam line	3##	~~15*~~ 19
5. Turbine Trip & Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	24
b. Steam Generator Water Level-High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2	19~~*~~
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

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
6. Auxiliary Feedwater (Continued)					
g. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to Essential Service Water)	~~2~~ 1/Train	~~2~~ 1/Train	~~2~~ 1/Train	1, 2, 3	15~~*~~ e
7. Automatic Opening of Containment Sump Suction Isolation Valves					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. RWST Level - Low-Low Coincident With Safety Injection	4	2	3	1, 2, 3, 4	~~16~~ 15
	See Item 1. above for Safety Injection initiating functions and requirements.				
8. Loss of Power					
a. ESF Bus Undervoltage	2/Bus	2/Bus	2/Bus	1, 2, 3, 4	25a~~*~~ e
b. Grid Degraded Voltage	2/Bus	2/Bus	2/Bus	1, 2, 3, 4	25b~~*~~ e

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
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Reactor Trip, P-4	4-2/Train	2/Train	2/Train	1, 2, 3	22
c. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
~~d. Steam Generator Water Level, P-14 (High-High)~~	~~4/stm. gen.~~	~~2/stm. gen. in any operating stm. gen.~~	~~3/stm. gen. in each operating stm. gen.~~	~~1, 2~~	~~20~~

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

#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 line pressure is not blocked.

~~*The provisions of Specification 3.0.4 are not applicable.~~

ACTION STATEMENTS

ACTION 14 - 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 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 ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within ~~1~~ 6 hours.

INSERT 3 →

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.

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.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

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~~ 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 4.3.2.1.

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 3.7.1.5.

ACTION 24 - 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 ~~2~~ 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 25 - a. With the number of OPERABLE channels one less than the Minimum Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. The inoperable channel may be bypassed for up to 2 hours for surveillance testing of the OPERABLE channel per Specification 4.3.2.1.

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

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, Start Diesel Generators, Containment Cooling Fans, Control Room Isolation, Phase "A" Isolation, Turbine Trip, Auxiliary Feedwater, Containment Vent Isolation and Essential 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-1	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low (Above P-11)	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low (Above P-11)	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-3	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 Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Vent Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	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
3.c. Containment Vent Isolation (Continued)								
2) Manual Phase "A" Isolation	See Item 3.a.1 above for all manual Phase "A" Isolation Surveillance Requirements.							
3) Manual Phase "B" Isolation	See Item 3.b.1 above for all manual Phase "B" Isolation Surveillance Requirements.							
4) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
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-2	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low (Above P-11)	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure - Negative Rate - High (Below P-11)	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	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

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
5. Turbine Trip and Feedwater (Continued)								
b. Steam Generator Water Level-High-High (P-14)	S	R	~~M~~ Q	N.A.	~~N.A.~~ M(1)	~~N.A.~~ M(1)	~~N.A.~~ Q	1, 2
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
6. 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 Relay	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. Undervoltage-RCP Bus	N.A.	R	N.A.	Q ~~M~~(3)	N.A.	N.A.	N.A.	1, 2
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Division 11 for Unit 1 (Division 21 for Unit 2) ESF Bus Undervoltage	N.A.	R	N.A.	M(2, 3)	N.A.	N.A.	N.A	1, 2, 3, 4
g. Auxiliary Feedwater Pump Suction Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. Automatic Opening of Containment Sump Suction Isolation Valves								

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. Automatic Opening (Continued)								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level-Low-Low	S	R~~#~~	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
Coincident With Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
8. Loss of Power								
a. ESF Bus Undervoltage	N.A.	R	N.A.	M(2, 3)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Grid Degraded Voltage	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Feature 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. Reactor Trip, P-4	N.A	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
c. Low-Low T_{avg}, P-12	N.A.	R~~#~~	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~d. Steam Generator Water Level, P-14 (High High)~~	~~S~~	~~R#~~	~~M~~	~~N.A.~~	~~M(1)~~	~~M(1)~~	~~Q~~	~~1, 2~~

TABLE NOTATION

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

(2) Undervoltage relay operability is to be verified independently. An inoperable channel may be bypassed for up to 2 hours for surveillance testing of the OPERABLE channel per Specification 4.3.2.1.

(3) Setpoint verification is not applicable.

~~# The specified 18 month interval may be extended to 32 months for Cycle 1 only.~~

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

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures 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 ~~is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.~~ Add Insert 1

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 safety 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. Add Insert 2

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + RE + SE \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for the actuation. RE or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. SE or Sensor Error is either

Insert #1:

New Bases Paragraph #1 (Add to existing paragraph.)

...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.

Insert #2:

New Bases Paragraph #2 (Add to existing paragraph.)

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 #3:

New Action to add to page 3/4 3-21

ACTION 15a - With the number of OPERABLE channels one less than the Total Number of Channels, declare the associated pump INOPERABLE and take the ACTION required by Specification 3.7.1.2.

Turbine Trip[#]

A Turbine trip initiates a Reactor trip. On decreasing power the Turbine trip is automatically blocked by ~~P-7 or~~ P-8 (a power level of approximately ~~10% (P-7) or~~ 30% (P-8) of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately ~~10% (P-7) or~~ 30% (P-8) of full power equivalent); and on increasing power, reinstated automatically by ~~P-7 or~~ 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.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position trips are anticipatory trips which provide core protection against DNB. The Open/Close Position trips assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Trip System. 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 more than one reactor coolant pump breaker is opened. Below P-7 the trip function is automatically blocked.

[#]~~A Reactor trip on Turbine trip is enabled above P-7 (10)% until the modification is implemented which enables Reactor trip on Turbine trip above P-8 (30%).~~

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 (i.e., prevents premature block of Source Range trip), provides an automatic backup block for Source Range Neutron Flux doubling, and the manual block that de-energizes the high voltage to the Source Range detectors. On decreasing power, Source Range Level trips and Neutron Flux doubling circuits 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, more than one reactor coolant pump breaker open, 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 and Turbine trip. On decreasing power, the P-8 automatically blocks the single loop low flow trip and Turbine trip.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range Reactor trip and the Low Setpoint Power Range Reactor trip; and automatically blocks the Source Range Reactor trip and provides an automatic backup function to de-energize 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 and Source Range high voltage to the detectors is restored if power decreases below the P-6 setpoint. Provides input to P-7.

P-13 Provides input to P-7.

~~A Reactor trip on Turbine trip is enabled above P-7 (10%) until the modification is implemented which enables Reactor trip on Turbine trip above P-8 (30%).~~

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					
a. High Setpoint	4	2	3	1, 2	2~~#~~
b. 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	5
7. Overtemperature ΔT	4	2	3	1, 2	6~~#~~
8. Overpower ΔT	4	2	3	1, 2	6~~#~~
9. Pressurizer Pressure-Low (Above P-7)	4	2	3	1	5~~#~~

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Pressurizer Pressure-High	4	2	3	1, 2	6#
11. Pressurizer Water Level-High (Above P-7)	3	2	2	1	6#
12. Reactor Coolant Flow-Low					
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 in 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#
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 (Above ~~P-7 or~~ P-8)~~****~~					
a. Emergency Trip Header Pressure	3/Train	2/Train	2/Train	1	6#
b. Turbine Throttle Valve Closure	4	4	1	1	6#

~~****A Reactor trip on Turbine trip is enabled above P-7 (10%) until the modification is implemented which enables Reactor trip on Turbine trip above P-8 (30%).~~

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*With the Reactor Trip System breakers in the closed position and the Control Rod Drive System capable of rod withdrawal.
~~**The boron dilution flux doubling signals may be blocked during reactor startup.~~
~~***These channels also provide inputs to ESFAS. The Action Statement for the channels in Table 3.3-3 is more conservative and, therefore, controlling.~~
~~#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.
@Whenever the Reactor Trip Bypass Breakers are racked in and closed for bypassing a Reactor Trip Breaker.

ACTION STATEMENTS

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

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

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

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

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

ACTION 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.

TABLE 3.3-1 (Continued)
ACTION STATEMENTS (Continued)

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 restore the inoperable channel to OPERABLE status within 48 hours or within the next hour open the reactor trip breakers, suspend all operations involving positive reactivity changes, and verify valves CV-111B, CV-8428, CV-8439, CV-8441 and CV-8435 are closed and secured in position. With no channels OPERABLE verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, and take the actions stated above within 1 hour and verify compliance 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.

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.

ACTION 9 - 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 ~~2~~ 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 ~~1~~ 6 hours.

ACTION 12 - a. With one of the diverse trip features (Undervoltage or Shunt Trip Attachment) inoperable, restore it to OPERABLE

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(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4) Q(4, 6), R(4, 5a)~~#~~	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)~~#~~	Q	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. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)~~#~~	Q	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5a)~~#~~	Q	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5b~~, 12~~)~~#~~	Q(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R(13)~~#~~	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R~~#~~	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low (Above P-7)	S	R~~#~~	Q~~**~~	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	P~~#~~	Q	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High (Above P-7)	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. Reactor Coolant Flow-Low	S	R#	Q	N.A.	N.A.	1
13. Steam Generator Water Level-Low-Low	S	R#	Q**	N.A.	N.A.	1, 2
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	Q**(10)	N.A.	1
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	Q(10)	N.A.	1
16. Turbine Trip (Above ~~P-7 or~~ P-8)***						
a. Emergency Trip Header Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Throttle 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 Coolant Pump Breaker Position Trip (Above P-7	N.A.	N.A.	N.A.	R	N.A.	1
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)#	~~Q~~ R	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)#	~~Q (8)~~ R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)#	~~Q (8)~~ R	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
19. Reactor Trip System Interlocks (Continued)						
d. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)#	~~Q (8)~~ R	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R#	~~Q (8)~~ R	N.A.	N.A.	1
20. Reactor Trip Breakers	N.A.	N.A.	N.A.	M (11)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	(15), R (16)	N.A.	1, 2, 3*, 4*, 5*

The initial single point comparison of incore to excore AXIAL FLUX DIFFERENCE following a refueling outage shall be performed prior to exceeding 75% of RATED THERMAL Power; Otherwise the

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

~~**These channels also provide inputs to ESFAS. The Operational Test Frequency for these channels in Table 4.3-2 is more conservative and, therefore, controlling.~~

~~#The specified 18 month interval may be extended to 32 months for cycle 1 only.~~

##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) ~~S~~single point comparison of incore to excore AXIAL FLUX DIFFERENCE shall be performed 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.

(5a) Initial plateau curves shall be measured for each detector. Subsequent plateau curves shall be obtained, evaluated and compared to the initial curves. 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.

(5b) With the high voltage setting varied as recommended by the manufacturer, an initial discriminator bias curve shall be measured for each detector. Subsequent discriminator bias curves shall be obtained, evaluated and compared to the initial curves.

(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.

Not used.

(8) ~~With power greater than or equal to the interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.~~

(9) 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. ~~Surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period.~~

(10) Setpoint verification is not applicable.

For the purposes of this surveillance, monthly shall mean at least once per 31 EFPD. The 24 hour completion time provisions of Specification 4.0.3 are not applicable.

For the purposes of this surveillance, quarterly shall mean at least once per 92 EFPD.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall be performed such that each train is tested at least every 62 days on a STAGGERED TEST BASIS and following maintenance or adjustment of the Reactor Trip Breakers and shall include independent verification of the OPERABILITY of the Undervoltage and Shut Trip Attachments of the Reactor Trip Breakers.

(12) Not used. ~~At least once per 18 months during shutdown verify that on a simulated Boron Dilution Doubling test signal CVCS valves 112D and E open and 112B and C close within 30 seconds.~~

(13) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

(14) Verify that the appropriate signals reach the Undervoltage and Shunt Trip relays, for both the Reactor Trip and Bypass Breakers from the Manual Trip Switches. ~~Initial performance of this surveillance requirement is to be completed prior to the Startup following the Unit 1 Cycle 1 Refuel Outage.~~

(15) Manual Shunt Trip prior to the Reactor Trip Bypass Breaker being racked in and closed by bypassing a Reactor Trip Breaker.

(16) Automatic undervoltage trip. ~~Initial performance of this surveillance requirement is to be completed prior to the startup following the Unit 1 Cycle 1 Refuel Outage.~~

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.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4 adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, either:

1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $Z + RE + SE \leq TA$

Where:

Z = The value from Column Z of Table 3.3-4 for the affected channel,

RE = The "as measured" value (in percent span) of rack error for the affected channel,

SE = Either the "as measured" value (in percent span) of the sensor error, or the value for Column SE (Sensor Error) of Table 3.3-4 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

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

~~*Control Room isolation not required prior to initial criticality on Cycle 1. Auxiliary Building Ventilation actuation not required prior to initial operation at ≥ 20% Rated Thermal Power (RTP) on Cycle 1.~~

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, Start Diesel Generators, Containment Cooling Fans, Control Room Isolation, Phase "A" Isolation, Turbine Trip, Auxiliary Feedwater, Containment Vent Isolation, and Essential 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-1	3	2	2	1, 2, 3	~~15*~~ 19*
d. Pressurizer Pressure-Low (Above P-11)	4	2	3	1, 2, 3#	19*
e. Steam Line Pressure-Low (Above P-11)	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#	~~15*~~ 19*
2. Containment Spray					
a. Manual Initiation	2 pair	1 pair	2 pair	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	4	2	3	1, 2, 3	16

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
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	23
2) System	2	1	2	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Containment Pressure-High-2	3	2	2	1, 2, 3	~~15*~~ 19*e
d. Steam Line Pressure-Low (above P-11)	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#	~~15*~~ 19*e
e. Steam Line Pressure - Negative Rate-High (below P-11)	3/steam line	2/steam line any steam line	2/steam line	3##	~~15*~~ 19*e
5. Turbine Trip & Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	24
b. Steam Generator Water Level-High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm gen.	3/stm. gen. in each operating stm. gen.	1, 2	19*e
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

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
6. Auxiliary Feedwater (Continued)					
g. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to Essential Service Water)	~~2~~ 1/Train	~~2~~ 1/Train	~~2~~ 1/Train	1, 2, 3	~~15*~~ 15a*
7. Automatic Opening of Containment Sump Suction Isolation Valves					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. RWST Level - Low-Low	4	2	3	1, 2, 3, 4	~~16~~ 15
Coincident With Safety Injection	See Item 1. above for Safety Injection initiating functions and requirements.				
8. Loss of Power					
a. ESF Bus Undervoltage	2/Bus	2/Bus	2/Bus	1, 2, 3, 4	25a*
b. Grid Degraded Voltage	2/Bus	2/Bus	2/Bus	1, 2, 3, 4	25b*

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
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Reactor Trip, P-4	4-2/Train	2/Train	2/Train	1, 2, 3	22
c. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
~~d. Steam Generator Water Level, P-14 (High High)~~	~~4/stm. gen.~~	~~2/stm. gen. in any operating stm. gen.~~	~~3/stm. gen. in each operating stm. gen.~~	~~1, 2~~	~~20~~

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

#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 line pressure is not blocked.

~~*The provisions of Specification 3.0.4 are not applicable.~~

restore the inoperable channel to OPERABLE status within 6 hours, or

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 the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 ~~2~~ hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

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

INSERT 3

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.

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.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

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~~ 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 4.3.2.1.

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 3.7.1.5.

ACTION 24 - 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 ~~2~~ 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 25 - a. With the number of OPERABLE channels one less than the Minimum Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. The inoperable channel may be bypassed for up to 2 hours for surveillance testing of the OPERABLE channel per Specification 4.3.2.1.

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

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, Start Diesel Generators, Containment Cooling Fans, Control Room Isolation, Phase "A" Isolation, Turbine Trip, Auxiliary Feedwater, Containment Vent Isolation and Essential 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-1	S	R	~~M~~ Q	N.A.	N.A.	N.A	N.A.	1, 2, 3
d. Pressurizer Pressure-Low (Above P-11)	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low (Above P-11)	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-3	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 Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R~~#~~	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Vent Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	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
3.c. Containment Vent Isolation (Continued)								
2) Manual Phase "A" Isolation	See Item 3.a.1 above for all manual Phase "A" Isolation Surveillance Requirements.							
3) Manual Phase "B" Isolation	See Item 3.b.1 above for all manual Phase "B" Isolation Surveillance Requirements.							
4) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
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-2	S	~~R~~ Q	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low (Above P-11)	S	~~R~~ Q	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure - Negative Rate - High (Below P-11)	S	~~R~~ Q	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	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

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
5. Turbine Trip and Feedwater (Continued)								
b. Steam Generator Water Level-High-High (P-14)	S	R~~#~~	~~M~~ Q	N.A.	~~N.A.~~ M(1)	~~N.A.~~ M(1)	~~N.A.~~ Q	1, 2
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
6. 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 Relay	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. Undervoltage-RCP Bus	N.A.	R	N.A.	~~M~~ Q(3)	N.A.	N.A.	N.A.	1, 2
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Division 11 for Unit 1 (Division 21 for Unit 2) ESF Bus Undervoltage	N.A.	R	N.A.	M(2,3)	N.A.	N.A.	N.A	1, 2, 3, 4
g. Auxiliary Feedwater Pump Suction Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. Automatic Opening of Containment Sump Suction Isolation Valves								

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. Automatic Opening (Continued)								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level-Low-Low Coincident With	S	R#	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
8. Loss of Power								
a. ESF Bus Undervoltage	N.A.	R	N.A.	M(2,3)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Grid Degraded Voltage	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Feature 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. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
c. Low-Low T_{avg}, P-12	N.A.	R#	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~d. Steam Generator Water Level, P-14 (High-High)~~	~~S~~	~~R#~~	~~M~~	~~N.A.~~	~~M(1)~~	~~M(1)~~	~~Q~~	~~1, 2~~

TABLE NOTATION

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

(2) Undervoltage relay operability is to be verified independently. An inoperable channel may be bypassed for up to 2 hours for surveillance testing of the OPERABLE channel per Specification 4.3.2.1.

(3) Setpoint verification is not applicable.

3/4.3.1 and 3/4.3.2 REACTOR TRIP S: STEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures 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 ~~is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.~~ Add INSERT 1

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 safety 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. Add INSERT 2

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + RE + SE \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for the actuation. RE or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. SE or Sensor Error is either

Insert #1:

New Bases Paragraph #1 (Add to existing paragraph.)

...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.

Insert #2:

New Bases Paragraph #2 (Add to existing paragraph.)

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 #3:

New Action to add to page 3/4 3-21

ACTION 15a - With the number of OPERABLE channels one less than the Total Number of Channels, declare the associated pump INOPERABLE and take the ACTION required by Specification 3.7.1.2.

ATTACHMENT 3A
EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS
WCAP 10271 AND EDITORIAL CHANGES

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards considerations 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.

Because these changes are generically applicable to the Chapter 15 analyses, the above attributes are being summarily addressed.

1. Does the change involve a significant increase the probability or consequences of an accident previously evaluated?

The determination that the results of the proposed change are within all acceptable criteria have been 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 References 1, 2 and 5. Implementation of the proposed changes is expected to result in an acceptable increase in total Reactor Protection System 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 Engineered Safety Features Actuation System 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) occurring during testing of RPS instrumentation. This reduction is primarily attributable to less frequent surveillance.

The reduction in inadvertent core melt frequency 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 analyses 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. The one plant-specific function evaluated on a plant specific basis for the Byron and Braidwood Nuclear Stations falls within the same criteria and is also considered to be acceptable.

The changes of an editorial nature have no impact on the severity or consequences of an accident previously evaluated.

Changes to Surveillance Test Frequencies for the Reactor Trip System 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 annunciator status light. 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 18 months does not therefore represent a significant change in channel surveillance and does not involve a significant increase in unavailability of the Reactor Protection System.

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 RPS but does not alter the manner in which protection is afforded nor the manner in which limiting criteria are established.

2.Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not involve hardware changes and do not result in a change in the manner in which the Reactor Protection System provides plant protection. No change is being made which alters the functioning of the Reactor Protection System. Rather the likelihood or probability of the Reactor Protection System functioning properly is affected as described above. Therefore the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3.Does the change 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:

a. Less frequent testing will result in less inadvertent reactor trips and actuation of Engineered Safety Features Actuation System components.

b. Higher quality repairs leading to improved equipment reliability due to longer repair times.

c. 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 Byron and Braidwood Nuclear Station Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety.

REFERENCES:

1. Letter from C. O. Thomas (NRC) to J. J. Sheppard (WOG) dated February 21, 1985 - "Safety Evaluation by the Office of Nuclear Reactor Regulation WCAP-10271, Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System".

2. Letter from Charles E. Rossi (NRC) to Roger A. Newton (WOG) dated February 22, 1989 - "Safety Evaluation by the Office of Nuclear Reactor Regulation Review of Westinghouse Report WCAP-10271 Supplement 2 and WCAP-10271 Supplement 2, Revision 1 on Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System".

3. WCAP-10271 Supplement 1-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", May 1986.

4. WCAP-10271-P-A Supplement 2, Revision 1 "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System", May 1989.

5. Letter Charles E. Rossi (NRC) to Gerard T. Goering (WOG) dated April 30, 1990 (NRC Supplemental Safety Evaluation for WCAP-10271 Supplement 2, Revision 1).

6. Technical Specification Optimization Program, RWST Switchover Justification for Byron and Braidwood Nuclear Stations.

ATTACHMENT 3B
EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS
AUXILIARY FEEDWATER PUMP SUCTION PRESSURE-LOW
(TRANSFER TO ESSENTIAL SERVICE WATER)

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards considerations 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.

Specification Table 3.3-3, Functional Unit 6.g, Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to Essential Service Water) Total Number of Channels, Channels to Trip, and Minimum Channels OPERABLE would be changed from two to one per train. The ACTION would also be changed such that if a channel were to become inoperable the associated AF pump would be declared inoperable and Specification 3.7.1.2 would be applied rather than placing the inoperable channel in the tripped condition and continuing operation until the performance of the next required ANALOG CHANNEL OPERATIONAL TEST.

The as-built plant configuration has only one suction pressure transmitter installed at the suction of each AF pump. A low suction pressure condition sensed by that transmitter in conjunction with an ESFAS actuation signal for its associated AF pump will initiate a transfer of the associated AF pump suction from the CST to the SX water supply. This actuation is train dependent and has a one-out-of-one actuation logic.

The current ACTION allows for continued operation until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within one hour. Rather than reducing the actuation logic to one-out-of-one, placing the channel in the tripped condition arms the transfer of the associated AF pump suction to the SX water supply. If the associated AF pump were to subsequently receive an ESFAS actuation signal, then the associated AF pump would start, its suction would be transferred to the SX water supply, and SX water would be injected into the steam generators. Injection of untreated SX water into the steam generators would have a devastating effect on secondary water chemistry and potentially shorten steam generator life. At a minimum, an extended outage would be required for secondary water chemistry cleanup and evaluation of long term effects. In order to preclude this potential event from occurring, current operating practice is to place the control switch for the associated AF pump in the pull out position rendering that pump inoperable prior to placing the inoperable AF pump suction pressure channel in the tripped condition. Specification 3.7.1.2 then becomes limiting requiring the associated AF pump to be restored to OPERABLE status within 72 hours or be in HOT STANDBY in the next six hours and in HOT SHUTDOWN within the following six hours.

The proposed ACTION would effectively impose a 72 hour allowed outage time (AOT) by invoking Specification 3.7.1.2. By not requiring the inoperable AF pump suction pressure channel to be placed in the tripped condition the control switch for the associated AF pump need not be placed in the pull out position leaving that pump available to manually or automatically respond in the event of an ESFAS actuation signal during the 72 hour AOT.

The effect on plant operation will be to increase the availability of the affected AF pump to respond manually or automatically to an ESFAS actuation signal by not requiring the inoperable AF pump suction pressure channel to be placed in the tripped condition. The remainder of the changes only serve to reflect the as-built plant configuration and mimic current plant operating practice.

This change will have no effect on reactivity management.

This change will not affect the failure of AF pump suction pressure channels. However, should a channel fail the associated AF pump will be declared inoperable without placing the inoperable channel in the tripped condition. If the channel was inoperable for reasons other than failing low, then the AF pump, although inoperable, would be available to respond manually or automatically in the event of an ESFAS actuation signal during the period of time the suction pressure channel is inoperable without injecting SX water into the steam generators.

The accidents that require AF system initiation are:

Inadvertent Opening of a Steam Generator Relief or Safety Valve,
Steam System Piping Failure,
Loss of External Load,
Loss of Non-emergency AC Power to the Plant Auxiliaries,
Loss of Normal Feedwater Flow,
Feedwater System Pipe Break,
Steam Generator Tube Rupture,
Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary, and
Anticipated Transients Without Scram.

The function of the AF system is the same for all accidents. The following is a generic discussion which applies to all accidents listed above.

The probability of an accident will not be increased by this change. This change is being made to accurately reflect the as-built plant configuration and to prevent an inadvertent injection of SX water into the steam generators should an AF pump suction pressure channel become inoperable.

The offsite dose consequences of previously analyzed accidents will not be increased. If an AF pump suction pressure channel was to become inoperable and its associated AF pump was also declared inoperable, then the other 100% capacity train of AF would be available to automatically respond to an ESFAS actuation signal to mitigate the consequences of these accidents. This is consistent with the initial assumptions of the accident analyses.

The probability of a malfunction of equipment important to safety is not affected by this changes. This change is being made to accurately reflect the as-built plant configuration. The probability of a failure of an AF pump suction pressure channel will not increase as result of this change. As a result of this change if an AF pump suction pressure channel was to become inoperable, then the associated AF pump will also be declared inoperable. This is consistent with the current operating practice that is required to ensure that SX water is not inadvertently injected into the steam generators when the inoperable AF pump suction pressure channel is placed in the tripped condition.

The consequences of a malfunction of equipment important to safety will not be increased. In the event that an AF pump suction pressure channel becomes inoperable and its associated AF pump is subsequently declared inoperable, the other 100% capacity train of AF would be available to automatically respond to an ESFAS actuation signal to mitigate the consequences of these accidents. Additionally, the revised ACTION limits the window of vulnerability during which an accident could occur while in this degraded condition to 72 hours. And, by not placing the inoperable channel in the tripped condition, the inoperable AF pump may, in some circumstances, remain available to help mitigate the consequences of these accidents.

The possibility of a new or different kind of accident or malfunction is not introduced. This change does not introduce any new plant equipment or require any installed plant equipment to be operated in a different manner. Declaring the affected AF pump inoperable has no impact on the initial assumptions of these accident analyses.

This change will not reduce the margin of safety. This change is being made to accurately reflect the as-built plant configuration and to prevent an inadvertent injection of SX water into the steam generators should an AF pump suction pressure channel become inoperable consistent with current operating practice of declaring the affected AF pump inoperable. Imposition of the 72 hour AOT for the AF pump will limit the window of vulnerability during which an accident could occur while in this degraded condition and will not reduce the margin of safety. The margin of safety will also not be reduced by not placing the inoperable AF pump suction pressure channel in the tripped condition since the other 100% capacity train of AF would be available to automatically respond to an ESFAS actuation signal to mitigate the consequences of these accidents and under certain circumstances the inoperable AF pump would also be available to respond manually or automatically to an ESFAS actuation signal.

ATTACHMENT 3C
EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS
RWST LEVEL CHANNEL CHANGE

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards considerations 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.

The proposed change will clarify the Action Statement for an inoperable RWST level channel. The current Action Statement requires an inoperable channel to be bypassed. Byron and Braidwood are configured such that an inoperable channel is removed from service by placing that channel in the tripped condition. In order to bypass the channel, temporary jumpers must be installed, or a circuit card removed. Testing in this configuration is contrary to IEEE 279 and the SER associated with WCAP 10271.

The new action statement will require RWST channels that are inoperable to be placed in a tripped condition within 6 hours. Operation may continue until the next required ANALOG CHANNEL OPERATIONAL TEST.

Each RWST is equipped with 4 level channels. This is one more than required by IEEE 279-1971, because there are no control functions associated with the subject channels. Other than alarm functions, these level channels provide an input to the SSPS. These inputs are associated with the semi-automatic switchover of the ECCS to the containment recirculation sumps. Upon reaching a level of 46.7% on two of the four level channels, concurrent with an SI signal, the containment sump suction valves will open. Manual action is then required to complete the realignment, which would isolate the RWST from the RH system. Prior to the completion of this realignment, the RH pumps take suction jointly from the containment sump and the RWST, i.e. the sump and the RWST are crosstied.

While a channel is being surveilled or is otherwise inoperable, it is placed in a tripped condition, consistent with the installed configuration. This results in a 1/3 coincidence concurrent with an SI signal to effect the opening of the containment sump suction valves. In this configuration, full compliance with IEEE 279-1971 is maintained.

This change has no effect on reactivity management.

For the period during which an RWST channel is inoperable, the failure of an additional RWST channel (channel fails low) will not result in the undesirable opening of the sump suction valves because an SI signal is also required. The affected modes are Modes 1 through 4, which coincides with the modes of applicability for the affected ECCS systems. Although an SI pump is required to be available in some Mode 5 and 6 configurations (during periods of reduced inventory operation), no credit for the semiautomatic switchover to the sump is assumed.

The accidents which result in an SI signal are:
Increased heat removal by the secondary,
Feedwater line break,
Spurious SI,
The range of LOCAs, and
Certain ATWS scenarios.

The limiting accident is the Large Break LOCA, which results in the greatest demand for RWST inventory, and thus results in the need to switch the RH pump suction to the containment recirculation sump at the earliest time. This accident bounds all other transients for the purposes of this change.

The probability of an accident will not be increased by this change. The large Break LOCA is analyzed assuming a full complement of ECCS equipment, with the subsequent failure of an entire train of this equipment. No allowance is made for the initiation of this transient with inoperable equipment, and it is recognized that the single failure criterion may not be met while operating under an Action Statement. The configuration used to remove inoperable RWST channels from service is unrelated to the probability that a catastrophic failure of the RCS piping will occur.

The offsite dose consequences of an accident are not increased by this change. As analyzed the Large Break LOCA does not result in unacceptable offsite dose consequences. In the event that a LOCA occurred with an RWST channel inoperable, the actuation logic to automatically open the sump suction valves would be one-out-of-three for the remaining operable RWST channels. This action would occur at the proper time assuming no failure of an additional RWST channel. It is recognized that single failure criterion cannot always be met when in an Action Statement due to already identified inoperable equipment.

The proposed revision will not increase the probability of a malfunction of equipment important to safety. No change is being made to installed plant equipment. The change is limited to method of removing an inoperable RWST Level channel from service.

The consequences of a malfunction of equipment important to safety is unchanged. This change does not render affected equipment vulnerable to a loss of suction, which could result in equipment failure. Multiple failures are required to result in the undesirable transfer of RWST inventory to the containment sump. The consequences of a loss of suction to the ECCS pumps due to multiple failures in the switchover circuitry are no worse than the consequences of a loss of suction to the ECCS pumps due to other causes, such as RWST catastrophic failure or personnel error. It must be reemphasized that the scenario leading to a loss of suction event requires multiple failures, which is beyond the design basis for the plant.

This change does not create the possibility for a new or different kind of accident or malfunction from those previously evaluated. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. Placing an inoperable RWST level channel in the tripped configuration does not render the plant vulnerable to a loss of suction which would result in equipment unavailability.

The margin of safety is not adversely impacted by the proposed change. The proposed change deals with the configuration of an inoperable RWST level channel. There is no change in the point at which a switchover of the ECCS pump suctions to the containment sumps is required. This change, as proposed, does not impact any analysis assumptions, and therefore, does not impact the analysis results. As such, the design margin of safety is unaffected.

ATTACHMENT 4
ENVIRONMENTAL ASSESSMENT

Commonwealth Edison Company has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration,

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.