REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINTS	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	≥90% of loop minimum measured flow*	≥89.3% of loop minimum measured flow*
13. Steam Generator Water Level Low-Low		
a. Unit 1	~~≥33.0% of narrow range instrument span~~ ≥33.0% (prior to cycle 9) ≥18.0% (cycle 9 and after) of narrow range instrument span	~~≥31.0% of narrow range instrument span~~ ≥31.0% (prior to cycle 9) ≥16.1% (cycle 9 and after) of narrow range instrument span
b. Unit 2	≥36.3% of narrow range instrument span	≥34.8% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	≥5268 volts - each bus	≥4920 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	≥57.0 Hz	≥56.08 Hz
16. Turbine Trip		
a. Emergency Trip Header Pressure	≥1000 psig	≥815 psig
b. Turbine Throttle Valve Closure	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.

*Minimum measured flow = 92,850 gpm# (97,600 gpm)**
**Not Applicable to Unit 1. Applicable to Unit 2 until completion of cycle 5.
#Applicable to Unit 1. Applicable to Unit 2 after cycle 5.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-2	≤8.2 psig	≤9.4 psig
d. Steam Line Pressure-Low (Above P-11)	≥640 psig*	≥614 psig*
e. Steam Line Pressure Negative Rate-High (Below P-11)	≤100 psi**	≤165.3 psi**
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)		
1) Unit 1	~~≤81.4% of narrow range instrument span~~ ≤ 81.4% (prior To Cycle 9) ≤ 88.0% (cycle 9 and after) of narrow range instrument Span	~~≤83.4% of narrow range instrument span~~ ≤ 83.4% (prior To cycle 9) ≤ 89.9% (cycle 9 and after) of narrow range instrument span.
2) Unit 2	≤80.8% of narrow range instrument span	≤82.8% of narrow range instrument span

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (continued)			
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.		
6. Auxiliary Feedwater			
a. Manual Initiation		N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays		N.A.	N.A.
c. Steam Generator Water Level-Low-Low-Start Motor-Driven Pump and Diesel-Driven Pump			
1) Unit 1		~~≥33.0% of narrow range instrument span~~ ≥ 33.0% (prior To cycle 9) ≥ 18.0% (cycle 9 and after) of narrow range instrument span	~~≥31.0% of narrow range instrument span~~ ≥ 31.0% (prior To Cycle 9) ≥ 16.1% (Cycle 9 and after) of narrow range instrument span
2) Unit 2		≥36.3% of narrow range instrument span	≥34.8% of narrow range instrument span
d. Undervoltage-RCP Bus-Start Motor Driven Pump and Diesel-Driven Pump		≥5268 volts	≥4920 volts
e. Safety Injection-Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.		

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant loops in operation when the Reactor Trip System breakers are closed and one reactor coolant loop in operation when the Reactor Trip System breakers are open:*

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

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

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

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

APPLICABILITY: MODE 3.**

ACTION:

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

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers

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

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side narrow range water level to be greater than or equal to ~~41% for Unit 1 (18% for Unit 2)~~ at least once per 12 hours.
18% (41% for Unit 1 prior to cycle 9)

4.4.1.2.3 The required coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

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

**See Special Test Exceptions Specification 3.10.4.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s) and/or RHR pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side narrow range water level to be greater than or equal to ~~41% for Unit 1 (18% for Unit 2)~~ at least once per 12 hours.
18% (41% for Unit 1 prior to Cycle 9)

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE#, or

b. The secondary side narrow range water level of at least two steam generators shall be greater than ~~41% for Unit 1 (18% for Unit 2)~~. 18% (41% for Unit 1 prior to cycle 9)

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator level as soon as possible.

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

##A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

ATTACHMENT B-1a

MARKED UP PAGES FOR PROPOSED CHANGES TO
APPENDIX A, IMPROVED TECHNICAL SPECFICATIONS
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Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
12. Undervoltage RCPs (per train)	$1^{(e)}$	4	J	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.14	≥ 4920 V
13. Underfrequency RCPs (per train)	$1^{(e)}$	4	J	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.14	≥ 56.08 Hz
14. Steam Generator (SG) Water Level - Low Low (per SG)					
a. Unit 1	1.2	4	D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.14	≥ 31.0% (Prior to Cycle 9) ≥ 16.1% (Cycle 9 and after)
b. Unit 2	1.2	4	D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.14	≥ 34.8%
15. Turbine Trip					
a. Emergency Trip Header Pressure (per train)	$1^{(f)}$	3	L	SR 3.3.1.10 SR 3.3.1.13	≥ 815 psig
b. Turbine Throttle Valve Closure (per train)	$1^{(f)}$	4	L	SR 3.3.1.10 SR 3.3.1.13	≥ 1% open
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1.2	2 trains	N	SR 3.3.1.12	NA
17. Reactor Trip Breakers[g]	1.2	2 trains	O	SR 3.3.1.4 SR 3.3.1.12	NA
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2 trains	R	SR 3.3.1.4 SR 3.3.1.12	NA

(continued)

(a) With Rod Control System capable of rod withdrawal or all rods not fully inserted.

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Above the P-8 (Power Range Neutron Flux) interlock.

(g) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.2-1 (page 4 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	$1,2^{(g)},3^{(g)}$	2 trains	I	SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.7	NA
b. Steam Generator (SG) Water Level - High High (P-14)					
1) Unit 1	$1,2^{(g)},3^{(g)}$	4 per SG	F	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.7 SR 3.3.2.10 SR 3.3.2.12	≤ 83.4% (Prior to Cycle 9) ≤ [illegible]% (Cycle 9 and after)
2) Unit 2	$1,2^{(g)},3^{(g)}$	4 per SG	F	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.7 SR 3.3.2.10 SR 3.3.2.12	≤ 82.8%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

(continued)

(g) Except when all required Feedwater Isolation Valves are closed or isolated by a closed manual valve

Table 3.3.2-1 (page 5 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	I	SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.7	NA
b. SG Water Level - Low Low				≥ 16.1% (Cycle 9 and after)	
1) Unit 1	1,2,3	4 per SG	F	SR 3.3.2.1 SR 3.3.2.6 SR 3.3.2.10 SR 3.3.2.12	≥ 31.0% (Prior to Cycle 9)
2) Unit 2	1,2,3	4 per SG	F	SR 3.3.2.1 SR 3.3.2.6 SR 3.3.2.10 SR 3.3.2.12	≥ 34.8%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
d. Loss of Offsite Power (Undervoltage on Bus 141(241))	1,2,3	2	F	SR 3.3.2.3 SR 3.3.2.10 SR 3.3.2.11	≥ 2730 V
e. Undervoltage Reactor Coolant Pump (per train)	1,2	4	Q	SR 3.3.2.8 SR 3.3.2.10 SR 3.3.2.12	≥ 4920 V
f. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	1 per train	M	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.10	≥ 2" Hg Vac
7. Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.7	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	D	SR 3.3.2.1 SR 3.3.2.6 SR 3.3.2.10 SR 3.3.2.12	≥ 44.7%
c. Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.5.2	Verify steam generator secondary side narrow range water level is ≥ 33% for each required Unit 1 RCS loop (≥ 37% for each required Unit 2 RCS loop).	12 hours
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to each required pump that is not in operation.	7 days

Prior to cycle 9 and ≥ 18% for cycle 9 and after

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.6.2	Verify SG secondary side narrow range water level is ≥ 33%, for each required Unit 1 RCS loop (≥ 37% for each required Unit 2 RCS loop).	12 hours
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to each required pump that is not in operation.	7 days

Prior to cycle 9 and ≥ 18% for cycle 9 and after

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify required RHR loop is in operation.	12 hours
SR 3.4.7.2	-------------------NOTE------------------- Only required when complying with LCO 3.4.7.b. -- Verify SG secondary side narrow range water level is ≥ 33% for each required Unit 1 RCS loop (≥ 37% for each required Unit 2 RCS loop).	12 hours
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to each required RHR pump that is not in operation.	7 days

Prior to cycle 9 and ≥ 18%
for cycle 9 and after

BASES

SURVEILLANCE REQUIREMENTS (continued)

<u>SR 3.4.5.2</u>

{Prior to Cycle 9 and ≥ 18% for cycle 9 and after}

SR 3.4.5.2 requires verification of required SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 33% for Unit 1 (≥ 37% for Unit 2) for each required RCS loop. If the SG secondary {Requirements not met} side narrow range water level is ~~< 33% for Unit 1 (< 37% for Unit 2)~~, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

<u>SR 3.4.5.3</u>

Verification that the required RCP is OPERABLE ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCP. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR, Section 15.4.1.

BASES

ACTIONS (continued)

C.1 and C.2

If no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RCS or RHR loop to OPERABLE status must be initiated. Boron dilution requires forced circulation to provide proper mixing and preserve the margin to criticality. The immediate Completion Times reflect the importance of maintaining the capability for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that the required operating RCS or RHR loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

Prior to Cycle 9 and ≥ 18% for Cycle 9 and after

SR 3.4.6.2

SR 3.4.6.2 requires verification of required SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 33% for Unit 1 (≥ 37% for Unit 2) for each required RCS loop. If the SG secondary side narrow range water level

Requirement is not met

is ~~< 33% for Unit 1 (< 37% for Unit 2)~~, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS

In MODE 5, RCS circulation increases the time available for mitigation of an accidental boron dilution event. The RHR loops provide this circulation and have been identified as important contributors to risk reduction.

RCS Loops – MODE 5, Loops Filled, satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs OPERABLE. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to provide adequate redundancy for heat removal. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels ≥ 33% for Unit 1 (≥ 37% for Unit 2) and with their associated RCS loops filled. Should the operating RHR loop fail, the SGs via natural circulation could be used to remove the decay heat.

Prior to Cycle 9 and ≥ 18% for Cycle 9 and after

Note 1 permits all RHR pumps to be removed from operation 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If no RHR loop is OPERABLE and one or both of the required SGs are inoperable, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status must be initiated. Boron dilution requires forced circulation to provide proper mixing and preserve the margin to criticality. The immediate Completion Times reflect the importance of maintaining the capability for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required operating RHR loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

Prior to cycle 9 and ≥ 18% for cycle 9 and after

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are ≥ 33% for Unit 1 (≥ 37% for Unit 2) ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level. This SR is modified by a Note which indicates that if both RHR loops are OPERABLE, this Surveillance does not need to be satisfied.

(continued)

ATTACHMENT B-2

MARKED UP PAGES FOR PROPOSED CHANGES TO
APPENDIX A TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-72 NPF-77

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REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	≥90% of loop minimum measured flow*	≥89.3% of loop minimum measured flow*
13. Steam Generator Water Level Low-Low		
a. Unit 1	≥33.0% of narrow range instrument span ≥ 33.0% (Prior To cycle 8) ≥ 18.0% (Cycle 8 and after) of narrow range instrument span	≥31.0% of narrow range instrument span ≥ 31.0% (Prior To Cycle 8) ≥ 16.1% (Cycle 8 and after) of narrow range instrument span
b. Unit 2	≥17% (Cycle 3); ≥36.3% (Cycle 4 and after) of narrow range instrument span	≥16.3% (Cycle 3); ≥34.8% (Cycle 4 and after) of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	≥5268 volts - each bus	≥4920 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	≥57.0 Hz	≥56.08 Hz
16. Turbine Trip		
a. Emergency Trip Header Pressure	≥1000 psig	≥815 psig
b. Turbine Throttle Valve Closure	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.

*Minimum measured flow = 97,600 gpm** (92,850 gpm)#
**Applicable to Unit 1 and Unit 2 until completion of cycle 5.
#Applicable to Unit 1 and Unit 2 starting with cycle 6.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-2	≤8.2 psig	≤9.4 psig
d. Steam Line Pressure-Low (Above P-11)	≥640 psig*	≥614 psig*
e. Steam Line Pressure Negative Rate-High (Below P-11)	≤100 psi**	≤165.3 psi**
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)		
1) Unit 1	~~≤81.4% of narrow range~~ instrument ~~span~~ ≤81.4% (prior to cycle 8) ≤88.0% (cycle 8 and after) of narrow range instrument span	~~≤83.4% of narrow range~~ instrument ~~span~~ ≤83.4% (prior to cycle 8) ≤89.4% (cycle 8 and after) of narrow range instrument span
2) Unit 2	≤78.1% (Cycle 3); ≤80.8% (Cycle 4 and after) of narrow range instrument span	≤79.7% (Cycle 3); ≤82.8% (Cycle 4 and after) of narrow range instrument span

TABLE 3.3-4 (continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (continued)			
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.		
6. Auxiliary Feedwater			
a. Manual Initiation		N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays		N.A.	N.A.
c. Steam Generator Water Level-Low-Low-Start Motor-Driven Pump and Diesel-Driven Pump			
1) Unit 1	≥ 33.0% (prior To Cycle 8) ≥ 18.0% (Cycle 8 and after) of narrow range instrument Span	≥33.0% of narrow range instrument span	≥31.0% of narrow range instrument span ≥ 31.0% (prior To Cycle 8) ≥ 16.1% (Cycle 8 and after) of narrow range instrument Span
2) Unit 2		≥17% (Cycle 3); ≥36.3% (Cycle 4 and after) of narrow range instrument span	≥16.3% (Cycle 3); ≥34.8% (Cycle 4 and after) of narrow range instrument span
d. Undervoltage-RCP Bus-Start Motor Driven Pump and Diesel-Driven Pump		≥5268 volts	≥4920 volts
e. Safety Injection-Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.		

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant loops in operation when the Reactor Trip System breakers are closed and one reactor coolant loop in operation when the Reactor Trip System breakers are open:*

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

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

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

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

APPLICABILITY: MODE 3.**

ACTION:

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

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

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

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side narrow range water level to be greater than or equal to ~~41% for Unit 1 (18% for Unit 2)~~ at least once per 12 hours.
18% (41% for Unit 1 prior to Cycle 8)

4.4.1.2.3 The required coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

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

**See Special Test Exceptions Specification 3.10.4.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s) and/or RHR pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side narrow range water level to be greater than or equal to ~~41% for Unit 1 (18% for Unit 2)~~ at least once per 12 hours.
18% (41% for Unit 1 prior To Cycle 5)

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE#, or

b. The secondary side narrow range water level of at least two steam generators shall be greater than ~~41% for Unit 1 (18% for Unit 2)~~. 18% (41% for Unit 1 prior to cycle 8)

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator level as soon as possible.

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

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*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

##A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

ATTACHMENT B-2a

MARKED UP PAGES FOR PROPOSED CHANGES
TO APPENDIX A IMPROVED TECHNICAL
SPECIFICATIONS OF FACILITY OPERATING
LICENSES
NPF-72, NPF-77

BRAIDWOOD STATION UNITS 1 & 2
REVISED PAGES,
3.3-16
3.3-35
3.3-36
3.4-10
3.4-13
3.4-16
B 3.4-29
B 3.4-34
B 3.4-37
B 3.4-40

Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
12. Undervoltage RCPs (per train)	1(e)	4	J	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.14	≥ 4920 V
13. Underfrequency RCPs (per train)	1(e)	4	J	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.14	≥ 56.08 Hz
14. Steam Generator (SG) Water Level - Low Low (per SG)					
a. Unit 1	1,2	4	D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.14	≥ 31.0%
b. Unit 2	1,2	4	D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.14	(Prior to cycle 5) ≥ 34.8% ≥ 16.1% (Cycle 5 and after)
15. Turbine Trip					
a. Emergency Trip Header Pressure (per train)	1(f)	3	L	SR 3.3.1.10 SR 3.3.1.13	≥ 815 psig
b. Turbine Throttle Valve Closure (per train)	1(f)	4	L	SR 3.3.1.10 SR 3.3.1.13	≥ 1% open
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	N	SR 3.3.1.12	NA
17. Reactor Trip Breakers(g)	1,2	2 trains	O	SR 3.3.1.4 SR 3.3.1.12	NA
	3(a), 4(a), 5(a)	2 trains	R	SR 3.3.1.4 SR 3.3.1.12	NA

(continued)

(a) With Rod Control System capable of rod withdrawal or all rods not fully inserted.

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Above the P-8 (Power Range Neutron Flux) interlock.

(g) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.2-1 (page 4 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	1,2$^{(g)}$,3$^{(g)}$	2 trains	I	SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.7	NA
b. Steam Generator (SG) Water Level - High High (P-14)					
1) Unit 1	1,2$^{(g)}$,3$^{(g)}$	4 per SG	F	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.7 SR 3.3.2.10 SR 3.3.2.12	≤ 83.4% (Prior to Cycle 8) ≤ 8?.9% (Cycle 8 and after)
2) Unit 2	1,2$^{(g)}$,3$^{(g)}$	4 per SG	F	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.7 SR 3.3.2.10 SR 3.3.2.12	≤ 82.8%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

(continued)

(g) Except when all required Feedwater Isolation Valves are closed or isolated by a closed manual valve.

Table 3.3.2-1 (page 5 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	I	SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.7	NA
b. SG Water Level - Low Low					
1) Unit 1	1,2,3	4 per SG	F	SR 3.3.2.1 SR 3.3.2.6 SR 3.3.2.10 SR 3.3.2.12	≥ 31.0% (Prior to Cycle 8) ≥ 16.1% (Cycle 8 and after)
2) Unit 2	1,2,3	4 per SG	F	SR 3.3.2.1 SR 3.3.2.6 SR 3.3.2.10 SR 3.3.2.12	≥ 34.8%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
d. Loss of Offsite Power (Undervoltage on Bus 141(241))	1,2,3	2	F	SR 3.3.2.3 SR 3.3.2.10 SR 3.3.2.12	≥ 2730 V
e. Undervoltage Reactor Coolant Pump (per train)	1,2	4	Q	SR 3.3.2.8 SR 3.3.2.10 SR 3.3.2.12	≥ 4920 V
f. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	1 per train	M	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.10	≥ 2" Hg Vac
7. Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.7	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	O	SR 3.3.2.1 SR 3.3.2.6 SR 3.3.2.10 SR 3.3.2.12	≥ 44.7%
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.5.2	Verify steam generator secondary side narrow range water level is ≥ 33% for each required Unit 1 RCS loop (≥ 37% for each required Unit 2 RCS loop).	12 hours
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to each required pump that is not in operation.	7 days

Prior to Cycle 8 and
≥ 18% for Cycle 8 and after

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.6.2	Verify SG secondary side narrow range water level is ≥ 33% for each required Unit 1 RCS loop (≥ 37% for each required Unit 2 RCS loop).	12 hours
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to each required pump that is not in operation.	7 days

Prior to Cycle 8 and ≥ 18% for cycle 8 and after

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify required RHR loop is in operation.	12 hours
SR 3.4.7.2	---------------------NOTE--------------------- Only required when complying with LCO 3.4.7.b. -- Verify SG secondary side narrow range water level is ≥ 33% for each required Unit 1 RCS loop (≥ 37% for each required Unit 2 RCS loop).	12 hours
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to each required RHR pump that is not in operation.	7 days

Prior to cycle 8 and ≥ 18% for cycle 8 and after

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.2

SR 3.4.5.2 requires verification of required SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 33% prior to Cycle 8 and ≥ 15% for Cycle 8 and after for Unit 1 (≥ 37% for Unit 2) for each required RCS loop. If the SG secondary side narrow range water level requirement is ~~< 33% for Unit 1 (< 37% for Unit 2)~~ not met, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

SR 3.4.5.3

Verification that the required RCP is OPERABLE ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCP. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR, Section 15.4.1.

BASES

ACTIONS (continued)

C.1 and C.2

If no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RCS or RHR loop to OPERABLE status must be initiated. Boron dilution requires forced circulation to provide proper mixing and preserve the margin to criticality. The immediate Completion Times reflect the importance of maintaining the capability for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that the required operating RCS or RHR loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

Prior to Cycle 8 and ≥ 15% for Cycle 8 and after

SR 3.4.6.2

SR 3.4.6.2 requires verification of required SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 33% for Unit 1 (≥ 37% for Unit 2) for each required RCS loop. If the SG secondary side narrow range water level is ~~< 33% for Unit 1 (< 37% for Unit 2)~~ Requirement not met, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS

In MODE 5, RCS circulation increases the time available for mitigation of an accidental boron dilution event. The RHR loops provide this circulation and have been identified as important contributors to risk reduction.

RCS Loops - MODE 5, Loops Filled, satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs OPERABLE. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to provide adequate redundancy for heat removal. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels ≥ 33% for Unit 1 (≥ 37% for Unit 2) and with their associated RCS loops filled. Should the operating RHR loop fail, the SGs via natural circulation could be used to remove the decay heat.

Prior to Cycle 8 and ≥ 18% for Cycle 8 and after

Note 1 permits all RHR pumps to be removed from operation 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If no RHR loop is OPERABLE and one or both of the required SGs are inoperable, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status must be initiated. Boron dilution requires forced circulation to provide proper mixing and preserve the margin to criticality. The immediate Completion Times reflect the importance of maintaining the capability for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required operating RHR loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

Prior to Cycle 8 and ≥ 18% for cycle 8 and after

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are ≥ 33% for Unit 1 (≥ 37% for Unit 2) ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level. This SR is modified by a Note which indicates that if both RHR loops are OPERABLE, this Surveillance does not need to be satisfied.

(continued)

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37, NPF-66, NPF-72, AND NPF-77

ComEd has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to Title 10 Code of Federal Regulations Section 50 Subsection 92 Paragraph c (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.

A. INTRODUCTION

Commonwealth Edison (ComEd) proposes to revise Byron and Braidwood Technical Specification (TS) Table 2.2-1 (functional unit 13.a), "Reactor Trip System Instrumentation Trip Setpoint: Steam Generator Water Level - Low-Low"; TS Table 3.3-4 (functional unit 5.b.1), "Engineered Safety Features Actuation System Instrumentation Trip Setpoint: Steam Generator Water Level - High-High"; TS Table 3.3-4 (6.c.1), "Engineered Safety Features Actuation System Instrumentation Trip Setpoint: Steam Generator Water Level - Low-Low Motor-Driven Pump and Diesel Driven Pump Start", TS Surveillance Requirement (TSSR) 4.4.1.2.2, required steam generator inventory during hot standby, TSSR 4.4.1.3.2, required steam generator inventory during hot shutdown, and TS Section 3.4.1.4.1.b, limiting condition for operation during cold shutdown with loops filled.

The installation of Babcock and Wilcox, International (BWI), replacement steam generators (RSGs) at the Byron Unit 1 and Braidwood Unit 1 Nuclear Power Stations necessitates an increase to the operating range of the steam generators due to the decrease

in narrow range span from 233 inches for the original Westinghouse Model D4 steam generators (OSGs) to 180 inches for the BWI RSGs. The increase in operating range will minimize the possibility of inadvertent plant trips following load changes and feedwater transients.

Commonwealth Edison also proposes to eliminate notations from page 2-5 for both Braidwood and Byron and pages 3/4 3-25 and 3/4 3-26 (for Braidwood only) since they are related to cycles already completed and, therefore, are no longer valid.

B. NO SIGNIFICANT HAZARDS ANALYSIS

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

This proposed change includes changing the low-low and high-high SG level setpoints. The setpoints are being changed to increase the SG level operating range. The change in acceptable operating range will decrease the possibility of inadvertent plant trips following load changes and feedwater transients. Therefore, the probability of inadvertent plant trips will decrease with this change.

The minimum setpoint change proposed in this request establishes controls to ensure that an adequate heat sink is maintained by providing an adequate secondary liquid mass to remove primary system sensible heat and core decay heat shortly after reactor trip and initiating auxiliary feedwater flow for long-term cooling. The accidents evaluated for this requirement are the Loss of Normal Feedwater and Feedwater Line Break transients.

The maximum setpoint ensures the steam lines and turbine remain undamaged from the introduction of low quality, two-phase flow from the steam generators into the steam lines. The accident evaluated for this requirement is the Feedwater System Malfunction that results in an increase in feedwater to one or more steam generators.

The steam generator water level setpoints are not considered a precursor to any of the analayzed accidents, and, therefore, these proposed changes do not result in an increase in the probability of occurrence of any accident previously analyzed.

The accidents evaluated for the low-low setpoint are the Loss of Normal Feedwater and Feedwater Line Break transients. These accidents were both analyzed using approved methodologies. All acceptance criteria were shown to be met for both these events. In addition, it was demonstrated that the Feedwater System Pipe Break response with the RSGs and the proposed low-low setpoint were bounded by the response with the original Model D4 steam generators. Therefore, the proposed low-low level setpoint change is demonstrated not to result in an increase in the consequences for these accidents.

The accident evaluated for the high-high setpoint is the Feedwater System Malfunction that results in an increase in feedwater to one or more Steam Generators. All acceptance

criteria were shown to be met. In addition, it was shown that the RSGs do not completely fill with liquid. This assures that the steam lines and turbine remain undamaged with no introduction of low quality, two-phase flow from the steam generators into the steam lines during the transient. With all acceptance criteria met, the proposed high-high level setpoint change is demonstrated not to result in an increase in the consequences for these accidents.

TSSR 4.4.1.2.2, TSSR 4.4.1.3.2, and TS 3.4.1.4.1.b assure a minimum inventory (i.e., level) to provide decay heat removal. The requirement for a minimum inventory to remove decay heat is met with assurance that the tube bundle is completely covered. The steam generator operating water level during shutdown conditions are not considered a precursor to any accident, and, therefore, these proposed changes do not result in an increase in the probability of occurrence of any accident previously analyzed.

The elimination of outdated cycle specific notations from page 2-5 for both Braidwood and Byron and pages 3/4 3-25 and 3/4 3-26 (Braidwood only) are only administrative and does not impact the probability or consequences of any accidents previously analyzed.

2. **The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed setpoint changes do not create any new operating conditions or modes. The proposed change only revises the setpoints for the Reactor Trip System and Engineered Safety Features Actuation System. The actions of these systems will continue to be performed in accordance with existing requirements which are sufficient to ensure plant safety is maintained.

Shutdown conditions steam generator water level is necessary to assure adequate decay heat removal capacity. Assurance that the tube bundle is completely covered along with existing technical specification controls on the Auxiliary Feedwater System and on the Condensate Storage Tank ensure adequate heat removal capacity is maintained and that plant safety is maintained.

Thus, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The elimination of outdated cycle specific notations from page 2-5 for both Braidwood and Byron and pages 3/4 3-25 and 3/4 3-26 (Braidwood only) are only administrative and does not create the possibility of a new or different accident.

3. **The proposed change does not involve a significant reduction in a margin of safety.**

A safety evaluation was performed to determine the effect of the RSGs with the revised setpoints.

The accidents potentially affected by the change in the Reactor Trip Steam Generator Water Level low-low setpoint (TS 2.2.1, Table 2.2-1, functional unit 13.a) and Engineered Safety Features Actuation System low-low AFW start setpoint (TS 3.3.2, Table 3.3-4, functional unit 6.c.1) are the Loss of Normal Feedwater and Feedwater Line Break transients. These accidents were both analyzed using approved methodologies. All acceptance criteria were shown to be met for both these events. In addition, it was demonstrated that the Feedwater System Pipe Break response with the RSGs with the proposed low-low setpoint were bounded by the response with the OSGs. Therefore, the proposed low-low level setpoint change is demonstrated not to result in an reduction in the margin of safety for these accidents.

The accident potentially affected by the change in the Engineered Safety Features Actuation System high-high SG level trip (TS 3.3.2, Table 3.3-4, functional unit 5.b.1) is a Feedwater System Malfunction that results in an increase in feedwater to one or more steam generators. This accident was analyzed using an approved methodology. In the evaluation of the Feedwater System Malfunction, all acceptance criteria were shown to be met. In addition, it was shown that the RSGs do not completely fill with liquid. This assures that the steam lines and turbine remain undamaged with no introduction of low quality, two-phase flow from the steam generators into the steam lines during the transient. With all acceptance criteria met, the proposed high-high level setpoint change is demonstrated not to result in a reduction in the margin of safety.

There are no design basis accidents involving shutdown condition steam generator water level. Existing TS controls on the Auxiliary Feedwater System and on the Condensate Storage Tank ensure adequate heat removal capacity is maintained and that plant safety is maintained during shutdown conditions. Therefore, a change to the shutdown condition steam generator water level does not result in a reduction in the margin of safety.

The elimination of outdated cycle specific notations from page 2-5 for both Braidwood and Byron and pages 3/4 3-25 and 3/4 3-26 (for Braidwood only) are only administrative and does not result in a reduction in the margin of safety for any analyzed event.

Therefore, this amendment request does not result in a significant decrease in a margin of safety.

Based on the above evaluation, ComEd has concluded that these changes involve no significant hazards considerations.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37, NPF-66, NPF-72, AND NPF-77

Commonwealth Edison Company (ComEd) has evaluated this proposed License Amendment Request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with Title 10, Code of Federal Regulations, Part 51, Section 21 (10 CFR 51.21). ComEd has determined that this proposed License Amendment Request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based upon the following.

1. The proposed licensing action involves the issuance of an amendment to a license for a reactor pursuant to 10 CFR 50 which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement. This proposed License Amendment Request changes the level setpoints for Byron and Braidwood Unit 1 steam generators after refueling outages B1R08 and A1R07, respectivily, due to the decrease in the narrow range span resulting from replacement of the original Westinghouse D4 steam generators with BWI steam generators.

2. This proposed License Amendment Request involves no significant hazards considerations as demonstrated in Attachment C;

3. There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite; and

4. There is no significant increase in individual or cumulative occupational radiation exposure.

Therefore, pursuant to 10 CFR 51.22(b), neither an environmental impact statement nor an environmental assessment is necessary for this proposed License Amendment Request.