ATTACHMENT B

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

BYRON UNITS 1 & 2	BRAIDWOOD UNITS 1 & 2
Revised Pages:	Revised Pages:
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*NOTE: THESE PAGES HAVE NO CHANGES BUT ARE INCLUDED FOR CONTINUITY.

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TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where: ΔT = Measured ΔT by RTD ~~Manifold~~ Instrumentation,

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

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s,

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

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3 = 0 s,# $\leq$ 2 s,##

ΔT_o = Indicated ΔT at RATED THERMAL POWER,

K_1 = 1.325*, (1.164)**

K_2 = 0.0297/°F*, (0.0265/°F)**

$\frac{1+\tau_4 S}{1+\tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation,

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 33 s, τ_5 = 4 s,

T = Average temperature, °F,

*Applicable to Unit 1. Applicable to Unit 2 after cycle 5.
**Not applicable to Unit 1. Applicable to Unit 2 until completion of cycle 5.
Applicable to Unit 1 until the completion of Cycle 7. Applicable to Unit 2 until the completion of cycle 6.
Applicable to Unit 1 Starting with cycle 8. Applicable to Unit 2 Starting with cycle 7.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

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

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s,[#] $\leq$ 2 s,[##]

T' $\leq$ 588.4°F (Nominal T_{avg} at RATED THERMAL POWER),

K_3 = 0.00181*, (0.00134)**

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, s^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between -24%* (-32%)** and +10%* (+13%)** $f_1(\Delta I)$ = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds +10%* (+13%)**, the ΔT Trip Setpoint shall be automatically reduced by 4.11%* (1.74%)** of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds -24%* (-32%)**, the ΔT Trip Setpoint shall be automatically reduced by 3.35%* (1.67%)** of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.16%* # (3.71%)** , 1.33% ## of ΔT span.

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*Applicable to Unit 1. Applicable to Unit 2 after cycle 5.

**Not applicable to Unit 1. Applicable to Unit 2 until completion of cycle 5.

#Applicable to Unit 1 until the completion of Cycle 7. Applicable to Unit 2 until the completion of cycle 6.

##Applicable to Unit 1 starting with cycle 8. Applicable to Unit 2 starting with cycle 7.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

Where:

ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 = 1.072,

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.00245/°F* (0.00170/°F)**, for T > T" and K_6 = 0 for T ≤ T",

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 588.4°F),

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.08%# (2.31%)** , 3.65%## of ΔT span.

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*Applicable to Unit 1. Applicable to Unit 2 after cycle 5.
**Not applicable to Unit 1. Applicable to Unit 2 until completion of cycle 5.
#Applicable to Unit 1 until completion of cycle 7. Applicable to Unit 2 until completion of Cycle 6.
##Applicable to Unit 1 Starting with cycle 8. Applicable to Unit 2 Starting with cycle 7.

Power Range, Neutron Flux, High Rates (Continued)

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

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor STARTUP to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. Both of these trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels in MODE 2 while the Source Range, Neutron Flux trip provides primary protection for the core in MODES 3, 4 and 5. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors ~~(about 4 seconds)~~, and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT Reactor trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

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

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

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

Pressurizer Water Level

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

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)	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-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 form 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)	R	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	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)	R	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	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*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

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

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

(1) If not performed in previous 31 days.

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

(3) 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 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. 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.

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

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

(8) Not used.

(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.

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.

(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.

(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.

* This note is applicable to Unit 1 until completion of Cycle 7 and Unit 2 until completion of Cycle 6

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power			
a. ESF Bus Undervoltage		2870 volts w/1.8s delay	≥2730 volts w/≤1.9s delay
b. Grid Degraded Voltage		3804 volts w/310s delay	≥3728 volts w/310 ± 30s delay
9. Engineered Safety Feature Actuation System Interlocks			
a. Pressurizer Pressure, P-11		≤1930 psig	≤1936 psig
b. Reactor Trip, P-4		N.A.	N.A.
c. Low-Low T_{avg}, P-12		≥550°F	≥547.2°F*, ≥ 546.9°F**
d. Steam Generator Water Level, P-14 (High-High)	See Item 5.b. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.		

* Applicable to Unit 1 until completion of cycle 7. Applicable to Unit 2 until completion of cycle 6.
** Applicable to Unit 1 Starting with cycle 8. Applicable to Unit 2 Starting with cycle 7.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where: ΔT = Measured ΔT by RTD ~~Manifold~~ Instrumentation,

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

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s,

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

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3 = 0 s,* $\le$ 2 s,**

ΔT_o = Indicated ΔT at RATED THERMAL POWER,

K_1 = 1.164,* 1.325**

K_2 = 0.0265/°F,* 0.0297/°F**

$\frac{1+\tau_4 S}{1+\tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation,

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 33 s, τ_5 = 4 s,

T = Average temperature, °F,

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* Applicable to Unit 1 and Unit 2 until completion of cycle 5.

** Applicable to Unit 1 and Unit 2 starting with cycle 6.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

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

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s*, ≤ 2s,**

T' ≤ 588.4°F (Nominal T_{avg} at RATED THERMAL POWER),

K_3 = 0.00134*, 0.00181**

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, s^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between -32%*, -24%** and +13%*, +10%** $f_1(\Delta I)$ = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds 13%*, +10%** the ΔT Trip Setpoint shall be automatically reduced by 1.74%*, 4.11%** of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds -32%*, -24%** the ΔT trip setpoint shall be automatically reduced by 1.67%*, 3.35%** of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.71%*, ~~1.16%**~~ 1.33%** of ΔT span.

*Applicable to Unit 1 and Unit 2 until completion of cycle 5.
**Applicable to Unit 1 and Unit 2 starting with cycle 6.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

Where:

ΔT	=	As defined in Note 1,
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	As defined in Note 1,
τ_1, τ_2	=	As defined in Note 1,
$\frac{1}{1 + \tau_3 S}$	=	As defined in Note 1,
τ_3	=	As defined in Note 1,
ΔT_o	=	As defined in Note 1,
K_4	=	1.072,
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation,
τ_7	=	Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.00170/°F*, 0.00245/°F** for T > T" and K_6 = 0 for T ≤ T",

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 588.4°F),

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.31%*, ~~3.08%**~~ 3.65%** of ΔT span.

*Applicable to Unit 1 and Unit 2 until completion of cycle 5.
**Applicable to Unit 1 and Unit 2 starting with cycle 6.

Power Range, Neutron Flux, High Rates (Continued)

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

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor STARTUP to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. Both of these trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels in Mode 2 while the Source Range, Neutron Flux trip provides primary protection for the core in Modes 3, 4 and 5. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors ~~(about 4 seconds)~~, and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT Reactor trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

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

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

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

Pressurizer Water Level

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

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)	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-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)	R	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	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)	R	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	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*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

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

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

(1) If not performed in previous 31 days.

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

(3) 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 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. For the purpose 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.

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

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

(8) Not Used.

(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.

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 Shut Trip Attachments of the Reactor Trip Breakers.

(12) Not Used.

(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.

(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.

* This note is applicable to Unit 1 and Unit 2 until completion of cycle 5.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE
8.	Loss of Power		
	a. ESF Bus Undervoltage	2870 volts w/1.8s delay	≥2730 volts w/≤1.9s delay
	b. Grid Degraded Voltage	3804 volts w/310s delay	≥3728 volts w/310 ± 30s delay
9.	Engineered Safety Feature Actuation System Interlocks		
	a. Pressurizer Pressure, P-11	≤1930 psig	≤1936 psig
	b. Reactor Trip, P-4	N.A.	N.A.
	c. Low-Low T_{avg}, P-12	≥550°F	≥547.2°F*, ≥546.9°F**
	d. Steam Generator Water Level, P-14 (High-High)	See Item 5.b. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.	

* Applicable to Unit 1 and Unit 2 until completion of Cycle 5.
** Applicable to Unit 1 and Unit 2 starting with cycle 6.

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

Commonwealth Edison (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

Early Westinghouse Pressurized Water Reactor (PWR) design of the Reactor Coolant System (RCS) used direct immersion Resistance Temperature Detectors (RTDs) to measure hot and cold leg temperatures. However, due to inadequate mixing of the coolant from the reactor core, the hot leg temperature was not uniform across the pipe cross-section (coolant temperature streaming). This problem did not exist at the cold leg location due to its proximity to the Reactor Coolant Pump (RCP) discharge.

Westinghouse developed an alternate method of measuring the coolant temperature of the hot and cold legs. This alternate design, currently installed at Braidwood and Byron is referred to as the RTD bypass system. The RTD bypass system directs flow from the hot leg through three flow scoops. The flow scoops are spaced equidistant around the circumference of the RCS hot leg and direct coolant through a common line to a manifold containing an RTD. Flow from the cold leg is directed to a manifold, also containing an RTD. These RTDs measure hot and cold leg temperatures, respectively. The flows from the hot and cold leg bypass piping unite downstream of the manifolds and flow into the crossover piping of the RCS at the suction of the RCP.

Although the RTD Bypass System reduced the temperature streaming problem, new problems were created. Among these problems is the increased maintenance due to leaking valves associated with the bypass piping, which has led to forced outages. Another problem is that the bypass piping is a significant crud trap and a significant source of radiation exposure when work is required on the system or other major components in the general area. Contact doses as high as 8 Rem/hr currently exist on some portions of the system. These high dose rates contribute to dose received during maintenance and in-service inspection of components in the system and during activities associated with nearby equipment (e.g., steam generators, reactor coolant pumps, loop isolation valves).

The proposed Technical Specification changes and associated plant modification will remove the RTD bypass piping on all four reactor coolant loops. The existing RTD bypass return line will be cut and capped at the reactor coolant crossover header. In place of the direct immersion single-element RTDs mounted in the manifolds, thermowell-mounted dual element fast response RTDs will be used. The hot leg scoops will be modified to accept new thermowells. The thermowell will be positioned to provide an average temperature reading for each scoop (the thermowell tip will be located at the third flow hole). A hole will be drilled through the end of each hot leg scoop to facilitate flow past the RTD. Water will enter through the existing flow holes, flow past the thermowell-mounted RTD, and exit through the new hole. The current cold leg RTD bypass penetration nozzle will also be modified to accept a thermowell-mounted dual element fast response RTD. A major benefit in using thermowell-mounted RTDs is that a faulty RTD may be replaced without breaching the RCS pressure boundary.

The new thermowell-mounted dual element fast response RTDs, manufactured by Weed Instrument Company, Inc. (model N9004E), will be placed in each of the three existing hot leg scoops and in the cold leg penetration of each loop. One element of each RTD will be active; the other will serve as an installed spare. The three hot leg temperature signals will be electronically averaged in the reactor protection system (RPS) to produce a representative hot leg temperature. This will necessitate the addition of a number of new cards to the 7300 Process Protection Cabinets, resulting in the removal of a two-tier card frame and the addition of a three tier card frame for each protection cabinet. The spare RTD element will be wired to the 7300 cabinets to facilitate switching to the spare element at the racks in the event of a failure of the active element.

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.

The proposed modification replaces the existing bypass piping system with thermowell-mounted RTDs. Because the hot leg RTDs are mounted directly in the scoops, temperature measurement inaccuracies caused by imbalances in the flow scoop sample flow are eliminated. The method of measuring coolant temperature with thermowell-mounted fast response RTDs has been analyzed to be at least as effective as the RTD bypass system. With the thermowells welded into the existing RCS hot and cold leg nozzles and the elimination of the bypass piping, the number of pressure boundary welds has been significantly reduced, resulting in a reduced probability of a small break LOCA.

The RTD response time is incorporated in the safety analyses. In particular, RTD response time is modeled in the OTΔT and OPΔT trip functions. The overall response time modeled in the safety analyses for the existing RTD bypass piping system is 8 seconds. The overall response time is the elapsed time from the time the temperature change in the RCS exceeds the trip setpoint until the rods are free to fall. More specifically, 6 seconds is modeled as a first order lag term and 2 seconds as pure delay on the reactor trip signal. The 6 second lag term includes such factors as: RTD bypass piping fluid transport delay, RTD bypass piping thermal lag, RTD response time, and RTD electronic filtering. The 2 second delay on reactor trip addresses such factors as electronics delay, trip breakers and gripper release.

Signal conditioning (filtering) of the individual loop ΔT and T_{avg} signals is represented by τ_3 and τ_6 , respectively, in the OTΔT and OPΔT equations in Technical Specification Table 2.2-1. With the current bypass manifold system, the filter is not required since the existing RTDs do not respond rapidly to local temperature variances within the reactor coolant loop. The bypass piping and manifold provide adequate mixing of the coolant, eliminating any local temperature variances. Therefore, the values of τ_3 and τ_6 are currently specified as 0 seconds, effectively turning off the electronic filter. The new fast response RTDs may respond to temperature spikes which are not representative of actual RCS bulk fluid temperature. Signal conditioning may be required to eliminate these temperature spikes. Although, the current Technical Specifications do not provide for any signal conditioning, the 8 second total response time used in safety analyses has sufficient margin to account for a typical 2 second time constant for signal conditioning. Industry experience has shown that a 2 second filter is adequate in eliminating the spikes.

The proposed fast response RTD/thermowell system also has an overall response time of 8 seconds. However, the time distribution for the parameters differ between the existing and proposed designs. The existing design includes a transport time for RCS fluid to reach the RTD, located in the manifold. The RTDs are directly immersed into the coolant, providing a fast response. The new design no longer has the transport delay. However, because the RTDs are mounted in thermowells, the response time of the RTD/thermowell combination will be increased over the existing system.

The effects of a redistribution of the time responses between the total lag term (pipe transport delay, RTD response and electronic filter delay) and electronics delay term have been evaluated. Westinghouse completed a Safety Evaluation SECL-95-015, "OTΔT and OPΔT Reactor Trip Response Time Safety Evaluation" to support the revision to the time requirements. The evaluation concludes that, as long as the total response time remains ≤ 8 seconds, the safety analyses acceptance criteria continue to be met. The OTΔT and OPΔT trip functions are unaffected by the change.

The following Updated Final Safety Analysis Report (UFSAR) Chapter 15 events trip on OTΔT: Loss of Electric Load/Turbine Trip, Uncontrolled RCCA Bank Withdrawal at Power, CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant, and Inadvertent Opening of a Pressurizer Safety or Relief Valve. In addition, the following events trip on OPΔT: Steamline Break at Hot Full Power for Core Response, and Steamline Break Superheat Analysis. These events have been reviewed for a change in the distribution of time responses for OTΔT and OPΔT. The review concludes that the time response redistribution did not result in a minimum DNBR lower than the safety analyses limit, did not result in a fuel centerline melt, nor did the superheated steam releases change from those currently existing. Therefore, the radiological consequences for these events do not increase as a result of the less restrictive time response breakdown. Thus, the proposed amendment does not result in an increase in the probability or consequences of a previously evaluated accident.

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

The OTΔT and OPΔT trip functions are unaffected by the change. Electronic filtering of the RTD signal has been included, changing the dynamic compensation term of OTΔT and OPΔT setpoint equations. No other changes to the setpoint equation result from the proposed modification.

The added 7300 hardware is compatible with the existing 7300 electronic hardware now used. All changes to the 7300 protection cabinets have been qualified. The proposed system is functionally equivalent to the existing one. The proposed modification has been reviewed for conformance with the Institute of Electrical and Electronics Engineers (IEEE) 279-1971 criteria, associated General Design Criteria, Regulatory Guides, and other applicable industry standards. The single failure criterion is satisfied by the proposed modification, since the independence of redundant protection sets is maintained. The new RTD/thermowell system meets the equipment seismic and environmental qualification requirements of IEEE standards 344-1975 and 323-1974, respectively. The proposed changes do not affect the protection system capabilities to initiate a reactor trip. The 2 of 4 voting coincidence logic of the protection sets is maintained. Therefore, the proposed modification meets all appropriate IEEE criteria, industry standards and other guidelines.

In addition, the RTD outputs are used for rod control, turbine runback, pressurizer level and other control systems. These control systems receive the signal after it has been processed at the 7300 cabinets and are therefore unaffected by the proposed modification.

The design and installation of the thermowells is in accordance with the American Society of Mechanical Engineers (ASME) Code requirements. However, should a thermowell fail at the RCS pressure boundary, the resulting accident is enveloped by current design basis accident analyses. Thus, implementation of the proposed amendment does not create the possibility of a new or different kind of accident from any of those previously evaluated.

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

The 7300 protection cabinets calculate individual loop ΔT and T_{avg}, based on the output of the RTDs. These values are used in the OTΔT and OPΔT reactor protection trip signals. Electronic filtering of the RTD signal will be included, changing the dynamic compensation term of OTΔT and OPΔT setpoint equations. No other changes to the setpoint equation result from the proposed modification. Although the total response time used as input into the safety analyses is unaffected by the proposed modification, the distribution of response times between the total lag (pipe transport delay, RTD response and electronic filter delay) and the electronic delay has changed. The UFSAR events which rely on OTΔT and OPΔT trips have been evaluated. The evaluation concludes that the safety analyses acceptance criteria continue to be met, since the total response time is consistent with the safety analyses. The OTΔT and OPΔT trips function in the same manner to terminate DNB-related transients. The reliability of the reactor protection system is unaffected by this change. Thus, the proposed modification does not involve a significant reduction in margin of safety.

Therefore, based on the above evaluation, Commonwealth Edison has concluded that these changes involve no significant hazards considerations.

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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 has evaluated the proposed amendment against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21. It has been determined that the proposed change meets the criteria for a categorical exclusion set forth in 10CFR51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10CFR50 and the amendment meets the following specific criteria:

(i) the amendment involves no significant hazards considerations

As demonstrated in Attachment C, this proposed amendment does not involve a significant hazards considerations.

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released off site

There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste. The proposed modification will have no impact on the types or quantities of radioisotope production. As such, the change will not affect the types or amounts of any effluents that may be released off site.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure

The proposed modification will likely result in significant reductions in occupational exposure due to the elimination of crud traps and the reduced maintenance associated with removal of the RTD bypass manifold system. Radiation exposure is accumulated not only in maintaining the RTD bypass manifold system, but in performing any work near the RTD bypass manifold system. Removal of the RTD bypass manifold system is expected to result in a radiation dose savings of approximately 52 person-rem per refueling cycle per unit. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

Commonwealth Edison has evaluated the proposed amendment against the criteria and found the changes meet the categorical exclusion permitted by 10CFR51.22(c)(9).

ATTACHMENT E

Westinghouse letter CAE-95-105/CCE-95-112, "Reactor Trip Response Time Safety Evaluation SECL-95-015", dated January 30, 1995.

Westinghouse Electric Corporation

Energy Systems

CAE-95-105
CCE-95-112
January 30, 1995

Mr. R. Belair
Commonwealth Edison Company
Braidwood Nuclear Station
Rural Route #1, Box 84
Braceville, IL 60407

Commonwealth Edison Company
Byron and Braidwood Units 1 & 2
Reactor Trip Response Time Safety Evaluation SECL-95-015

Reference: Byron/Braidwood RTD Bypass Elimination Program (ComEd P.O. 341716 - YY80C)

Dear Mr. Belair:

Attached is Safety Evaluation SECL-95-015, "OTΔT/OPΔT Reactor Trip Response Time Safety Evaluation," to support a revision to the response time requirements for the OTΔT and OPΔT reactor trip functions at Byron and Braidwood. This safety evaluation was requested by ComEd to support ComEd's RTD Bypass Elimination program licensing submittal schedule. There are no Technical Specification changes associated with this evaluation. The response time requirements for the OTΔT and OPΔT reactor trip functions are in Chapter 16 of the Byron/Braidwood Updated FSAR. A proposed revision to FSAR Table 16.3-1 is attached to this evaluation. Updates to the PLS are not part of this effort. (Any PLS updates required are ComEd responsibility, consistent with the scope of the RTD Bypass Elimination Program.

0368t:1/30/95

This safety evaluation is based on the latest Byron/Braidwood licensing basis safety analyses supporting the Reduced TDF and Increased SGTP program, WCAP-13964, Revision 2. As such, it is applicable to all four units once the WCAP-13964 analyses become effective. Byron 1, beginning with Cycle 7 and Byron 2, beginning with Cycle 6, both apply the WCAP-13964 licensing basis analyses. The Braidwood units are both expected to use the WCAP-13964 licensing basis analyses after their next respective refueling outages. For Braidwood, this timing is consistent with planned operation with the RTD Bypass System eliminated. The revised response time requirements will also be considered in the safety evaluations for Byron/Braidwood being performed to support operation with an average SGTP level of 24% with a maximum 30% SGTP level in any one steam generator.

If you have any questions, please contact us.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

Jand M Buncicky for

B. S. Humphries, Manager
Commonwealth Edison Project
Operating Plant Programs

cc:	T. O'Connor	ComEd/NFS
	K. Kovar	ComEd/NFS
	S. Ahmed	ComEd/NFS
	D. St. Clair	ComEd/Byron
	R. Kerr	ComEd/Braidwood
	P. Reister	ComEd/Byron
	P. H. McHale	Westinghouse/Byron

WESTINGHOUSE NUCLEAR SAFETY
SAFETY EVALUATION CHECK LIST (SECL)

1.) NUCLEAR PLANT(S) Byron & Braidwood Units 1 & 2 - CAE CBE CCE CDE

2.) SUBJECT (TITLE): OTΔT/OPΔT Reactor Trip Response Time Safety Evaluation

3.) The written safety evaluation of the revised procedure, design change or modification required by 10 CFR 50.59(b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A - 10 CFR 50.59(a)(1)

3.1) Yes X No __ A change to the plant as described in the FSAR?

3.2) Yes __ No X A change to procedures as described in the FSAR?

3.3) Yes __ No X A test or experiment not described in the FSAR?

3.4) Yes __ No X A change to the plant Technical Specifications? (See Note on Page 2.)

4.) CHECK LIST - PART B - 10 CFR 50.59(a)(2) (Justification for Part B answers must be included on page 2.)

4.1) Yes __ No X Will the probability of an accident previously evaluated in the FSAR be increased?

4.2) Yes __ No X Will the consequences of an accident previously evaluated in the FSAR be increased?

4.3) Yes __ No X May the possibility of an accident which is different than any already evaluated in the FSAR be created?

4.4) Yes __ No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

4.5) Yes __ No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

4.6) Yes __ No X May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

4.7) Yes __ No X Will the margin of safety as described in the bases to any Technical Specification be reduced?

NOTES:

If the answer to any of the above questions is unknown, indicate under 5.) REMARKS and explain below.

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written safety evaluation, the change review would require an application for license amendment as required by 10 CFR 50.59(c) and submitted to the NRC pursuant to 10 CFR 50.90.

5.) REMARKS:

The answers given in Section 3, Part A, and Section 4, Part B, of the Safety Evaluation Checklist, are based on the attached Safety Evaluation.

See following 7 pages.

FOR FSAR UPDATE

Table 16.3-1 (Page 16.3-2) requires revision. The proposed revision is attached.

Reason for/Description of Change:

Table 16.3-1 of the FSAR requires a change to reflect the new OTΔT / OPΔT reactor trip response time requirements supported by this evaluation.

6.) SAFETY EVALUATION APPROVAL LADDER:

NS Preparer: Gary G. Ament Date: 1/30/95

NS Reviewer: G. L. Sharp Date: 1/30/95

NS Group Manager: John Check for CSH C. S. Hauser Date: 1/30/95

OPL Engineer R. J. Morrison Date: 1/30/95

OTΔT / OPΔT REACTOR TRIP RESPONSE TIME SAFETY EVALUATION FOR BYRON / BRAIDWOOD

1.0 BACKGROUND AND INTRODUCTION

The Overtemperature ΔT (OTΔT) reactor trip function provides primary protection against departure from nucleate boiling (DNB) during postulated transients in Westinghouse reactors. The indicated ΔT is used as a measure of reactor power and is compared with a setpoint that is automatically varied, depending on Tavg, pressurizer pressure, and axial flux difference (ΔI). If the OTΔT signal exceeds the calculated setpoint in two or more channels the reactor is tripped.

The Overpower ΔT (OPΔT) reactor trip function is designed to protect against a high fuel rod power density and subsequent fuel rod cladding failure and fuel melt. The indicated ΔT is used as a measure of reactor power and is compared with a setpoint that is automatically varied depending on Tavg. If the OPΔT signal exceeds the calculated setpoint in two or more channels the reactor is tripped.

The ΔT that is compared to the OTΔT and OPΔT setpoints is calculated from the RTDs which measure the hot and cold leg temperature. The indicated Tavg, which is also calculated based on the hot and cold leg temperature measured by the RTDs, is also an input to the OTΔT and OPΔT setpoint equations. Therefore, the response of these trip functions is dependent on the measurement system of the hot and cold leg temperatures. The current plant configuration for the Byron/Braidwood units consists of a bypass loop with the RTDs mounted in the manifold of the bypass loop. Due to potential radiological concerns associated with maintenance of the RTDs in the bypass loops, Commonwealth Edison plans to remove the RTD bypass loops and replace them with RTDs mounted in thermowells. The existing scoops in the hot and cold leg reactor coolant loop piping will be modified and the RTD thermowells will then be located directly in these scoops. The program to support this change is the RTD Bypass Elimination program.

For both the existing system and the RTD Bypass Elimination configuration, the OTΔT and OPΔT reactor trip model used in the safety analysis includes a 6 second first order lag for the temperature sensor response and a 2 second delay for the electronic response, for a combined total time of 8 seconds from the time that the setpoint is reached to the loss of stationary gripper coil voltage and the RCCAs are free to fall. The first order lag for the RTD is used since the RTD behaves as a first order device as opposed to a pure time delay. The 2 second electronic delay is a pure time delay. Due to the first order lag, the total response time of the system is dependent on the nature of the transient, i.e., there is no single time that represents the response time of the system. To verify that the system operates in a manner consistent with the safety analyses, the RTD response time and any filter type delays that function as a lag must be $\leq$ 6 seconds and the electronic delay must be $\leq$ 2 seconds. Due to the different nature of these responses, these two times are not additive. In addition, due to physical constraints and existing surveillance procedures at Byron/Braidwood, the response time testing required to show compliance with the safety analysis response time assumptions for the OTΔT and OPΔT trip functions requires multiple measurements and manipulations of the test results. To support simplification of the testing procedures and to take credit for the fact that pure delay component of the overall response time is usually well below 2 seconds, margin available in the applicable safety analyses that credit reactor trip on these functions can be used to support combining the RTD response time (lag) and pure delay, regardless of the distribution within the combined total of 8 seconds. This is the purpose of this safety evaluation.

The table below shows the breakdown of times for the existing system model and the proposed surveillance criteria. The proposed criteria would allow a combination of first order lag (RTD response time) and electronic delay of up to 8 seconds with no constraint on either the first order lag or the pure electronic delay response time.

	Existing Surveillance Criteria	Proposed Surveillance Criteria
First Order Lag	≤ 6.0 sec	NA
Pure Delay	≤ 2.0 sec	NA
Total Response Time	NA	≤ 8.0 sec

Table 15.0-5 of the Byron/Braidwood Update FSAR (Reference 1) provides response times for reactor trips. This table currently reflects the total overall requirement of 8 seconds for the OTΔT and OPΔT trip functions. Table 16.3-1 of the FSAR provides acceptance criteria for response time tests. The proposed revisions to this table which are attached to this safety evaluation indicate the limits that the measured response times and delay times must fall within. The evaluation presented below demonstrates that the proposed surveillance limits will assure that the results of the safety analyses remain valid.

2.0 LICENSING BASIS

In order to allow a combined total RTD response time and electronic delay of ≤8 seconds, regardless of the distribution of each within this total time, it is necessary to assess the impact of the changed OTΔT and OPΔT response times on the accidents that rely on those trips for protection. The following Byron/Braidwood FSAR events trip on OTΔT:

1. Loss of Electrical Load / Turbine Trip (FSAR Sections 15.2.2 and 15.2.3)

2. Uncontrolled RCCA Bank Withdrawal at Power (FSAR Section 15.4.2)
3. CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (FSAR Section 15.4.6)
4. Inadvertent Opening of a Pressurizer Safety or Relief Valve (FSAR Section 15.6.1)

The following Byron/Braidwood analyses trip on OPΔT:

1. Steamline Break at Hot Full Power for Core Response
2. Steamline Break Superheat Analysis

3.0 EVALUATION

The impact on the above accidents due to the change in the OTΔT and OPΔT response time breakdown will be addressed in this section.

Loss of Electrical Load / Turbine Trip

The Loss of Electrical Load / Turbine Trip transients rely upon the High Pressurizer Pressure and OTΔT reactor trip functions for protection. For these events, four cases are considered. Two cases are analyzed for peak RCS pressure considering maximum and minimum reactivity feedback conditions. These two cases trip on High Pressurizer Pressure and, hence, are not effected by the subject change in the OTΔT/OPΔT response times. The other two cases, which also consider maximum and minimum reactivity feedback conditions, are analyzed for minimum DNBR. Of these two cases, only the maximum reactivity feedback case trips on OTΔT. Due to the anticipatory nature of the OTΔT trip function (i.e., lead/lag compensation), the minimum DNBR never falls below the initial value in this case as shown in FSAR Figure 15.2-4. Since the initial DNBR value is well above the safety analysis limit value (i.e., approximately 52% DNBR margin) and is not a limiting concern for these conditions, the safety analysis DNBR limit will continue to be met considering the change in the distribution of the lag and delay time components for the OTΔT trip function. Hence, the conclusions presented in the FSAR for these events remain valid.

Uncontrolled RCCA Bank Withdrawal at Power

This transient relies upon OTΔT and High Neutron Flux for protection to ensure that the minimum DNBR remains above the limit value. A spectrum of reactivity insertion rates is considered from several initial power levels to demonstrate that these two trips provide adequate protection. These analyses are performed assuming both minimum and maximum reactivity feedback. The cases that trip on high neutron flux will be unaffected by the change in response time breakdown while rod motion may be delayed slightly for those cases that trip on OTΔT.

As can be seen in Figures 15.4-8 through 15.4-10 of the Byron/Braidwood FSAR, OTΔT provides the primary protection for lower reactivity insertion rates. Using the results of existing analyses for this event that were performed to determine the effect of changes in the OTΔT response time breakdown, an evaluation was performed for Byron/Braidwood. In the existing analyses, several cases with maximum and minimum reactivity feedback at full power were analyzed with the time response breakdown varied over its full possible range of values. In addition, a 60% power and maximum reactivity feedback case was analyzed to include the effects of reduced power. Based on the results of the existing analyses and the evaluation for Byron/Braidwood, it has been shown that in all cases, the minimum DNBR remains above the safety analysis DNBR limit demonstrating that the conclusions presented in the FSAR remain valid.

Uncontrolled Boron Dilution

A change in OTΔT response time for the Boron Dilution events presented in the FSAR would only potentially affect the case analyzed at full power with manual rod control. In this event, the operator action time is measured from the time of reactor trip on OTΔT from the full power, minimum feedback, 0.6 pcm/sec insertion rate Rod Withdrawal at Power (RWAP) event. The RWAP sensitivity analysis showed that the time of rod motion in the RWAP event varied only slightly (< 1 second) for the various response time breakdowns. The time interval from the time of OTΔT trip to loss of required minimum shutdown margin calculated for the Uncontrolled Boron Dilution event at full power with manual rod control is 25.4 minutes. Hence, a change of <1 second is insignificant and well within the margin between the operator response time and loss of shutdown margin for the analysis of this event. Hence, the 15 minute criterion for operator action continues to be met and the conclusions presented in the FSAR remain valid.

Accidental RCS Depressurization

The RCS Depressurization event is analyzed to show that the minimum DNBR remains above the applicable safety analysis limit value. Using the results of existing analyses for this event that were performed to determine the effect of changes in the OTΔT response time breakdown, an evaluation was performed for Byron/Braidwood. In the existing sensitivity analyses for this event, the effect of various RTD response time breakdowns on the time of OTΔT reactor trip and the calculated minimum DNBR was examined. These sensitivity analyses show that minimum DNBR decreases as the pure delay portion of the overall response time increases to a pure delay of 8 seconds (0 second lag) and the maximum DNBR decrease is approximately 3% from the minimum DNBR for this event. DNBR as a function of time is illustrated in FSAR Figure 15.6-1 for this event and the minimum DNBR is approximately 30% above the current safety analysis limit DNBR. Hence, the net effect is a reduction in DNBR margin of 3.4% for this event and the resulting DNBR remains approximately 26.6% above the applicable safety analysis limit DNBR. Therefore, the safety analysis DNBR limit continues to be met and the conclusions of the FSAR licensing basis analysis for this event remain valid.

Steamline Break Core Response at Hot Full Power

Although not reported as part of the Byron/Braidwood licensing basis in FSAR Section 15.1.5, Steamline Break analyses for core response from hot full power conditions have been considered in previous evaluations for Byron/Braidwood. Initially, the Steamline Break analysis for core response at hot full power were considered in 1985 to support an evaluation addressing an increase in the RTD response time from 4 seconds to the current value of 6 seconds. Most recently, this event was reanalyzed for the evaluations supporting the reduction in TDF and increased SGTP analysis as reported in WCAP-13964, Revision 2 (Reference 2). In the analysis performed, the cases for steamline breaks from 0.4 ft^2 to 1.0 ft^2 trip on OPΔT protection. The most limiting conditions occur for the 1.0 ft^2 break case and subsequent evaluations of the resulting statepoint conditions showed that the DNB design basis and fuel centerline melt criteria was met. For the subject evaluation supporting a combined RTD response time (lag) and pure delay totalling 8 seconds, regardless of the distribution within the combined total, this limiting case was evaluated. The results of this evaluation show that the DNB design basis and fuel centerline melt criteria continues to be met. Hence, the conclusions of WCAP-13964 for this event remain valid.

Steamline Break Superheat Analysis

The superheated steam releases outside containment applicable for Byron/Braidwood are documented in letter CAE-88-240/CCE-88-321, "Revised Mass and Energy Release Data for Feedwater Bypass Check Valve Removal," Reference 3. The superheated steam release analysis has 3 cases which trip on OPΔT: Cases 5, 6, and 7. These cases trip early in the transient (before 30 seconds) compared to the total length of the event (1800 seconds). Based on sensitivities performed for the RCS Depressurization and RWAP events, it has been shown that the change in the distribution of the lag and pure delay, within the same total combined response time, delays a reactor trip on OTΔT by no more than a few seconds. Since the OPΔT is also a function of the same ΔT and Tavg used in the OTΔT trip function, the same magnitude of delay would be expected for the OPΔT trip function. A review of Tables B-5, B-6, and B-7 of the aforementioned letter shows that the maximum superheat values begin to occur after 400 seconds into the transient and remain for the duration of the analysis (1800 seconds). Hence, a rod motion delay of a few seconds in the superheat analysis for these cases will not result in any significant change in the overall profile of the mass and energy releases or superheat conditions due to the extended length of the transient. Therefore, the results of the superheated steam release analysis remain valid.

4.0 Determination of an Unreviewed Safety Question

The following responses are provided with respect to the Byron/Braidwood safety analyses.

4.1 Will the probability of an accident previously evaluated in the SAR be increased?

There are no accidents which would be more likely to occur due to a change in the OTΔT/OPΔT response time breakdown since the protection system response time and the normal operation of the plant do not impact each other.

4.2 Will the consequences of an accident previously evaluated in the SAR be increased?

A change in the OTΔT/OPΔT response time breakdown did not result in a minimum DNBR lower than the safety analysis limit, did not result in fuel centerline melt, nor did the superheated steam releases change from those currently existing. Therefore, the radiological consequences for these events do not increase as a result of the less restrictive response time breakdown.

4.3 May the possibility of an accident which is different than any already evaluated in the SAR be created?

A change in the response time breakdown will not initiate an accident which is not already considered in the FSAR. The specific response time values in no way affect the possibility that any event will occur.

4.4 Will the probability of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

This evaluation is in support of a less restrictive response time breakdown for a previously evaluated and approved equipment change. There is no known mechanical or electrical impact on the equipment due to the less restrictive response time breakdown which would increase the probability of the equipment to malfunction.

4.5 Will the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

There are no changes in the safety analysis results and conclusions which would impact radiological consequences. The evaluation has shown that the reactor core and resulting superheated steam releases are not adversely impacted by the proposed response time breakdown.

4.6 May the possibility of a malfunction of equipment important to safety different than any already evaluated in the SAR be created?

There is no known change in the OTΔT and OPΔT protection system equipment which would cause the malfunction of safety equipment assumed in the licensing basis safety analyses for Byron/Braidwood. No new mode of failure is created by the proposed response time breakdown.

4.7 Will the margin of safety as defined in the BASES to any technical specifications be reduced?

The margin of safety as defined in the BASES is not reduced in any accidents due to the proposed response time breakdown. The safety analysis acceptance criteria remain unchanged and continue to be met. Therefore, the same margin exists to the design failure point or system limitation and, hence, the margin to safety is not impacted.

5.0 CONCLUSIONS

Based on existing sensitivity studies and evaluations performed for all of the Byron/Braidwood licensing basis events which explicitly rely on the OTΔT and OPΔT reactor trips for protection, it is demonstrated that for a combined RTD response time (lag) and pure delay totalling 8 seconds, regardless of the distribution within the combined total, the DNBR safety analysis limits and other applicable safety analysis criteria continues to be met. The evaluations also show that due to the magnitude of the duration of the event, when compared to magnitude of an increase in the time of reactor trip on an OPΔT signal which may occur as a result of the change in the RTD lag and time delay distribution with a total combined time of 8 seconds, the results of the Steamline Break Outside Containment Superheat analysis applicable to Byron/Braidwood will not be affected by the proposed response time breakdown for this reactor trip function. Hence, the conclusions in the FSAR and supporting analysis basis documentation (i.e., WCAP-13964 and letter CAE-88-240/CCE-88-321) remain valid for all events which rely on the OTΔT and OPΔT reactor trips for protection.

6.0 REFERENCES

1. Byron/Braidwood Stations, Updated Final Safety Analysis Report, Chapters 15 and 16

2. WCAP-13964, Revision 2, "Byron and Braidwood Units 1 and 2 - Increased SGTP / Reduced TDF / PMTC Analysis Program - Engineering / Licensing Report," September 1994, T. J. Gerlowski, R. J. Morrison

3. CAE-88-240/CCE-88-321, "Revised Mass and Energy Release Data for Feedwater Bypass Check Valve Removal," J. L. Tain (Westinghouse) to Mr. D. Elias (ComEd), June 9, 1988

TABLE 16.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT		RESPONSE TIME
1.	Manual Reactor Trip	N.A.
2.	Power Range, Neutron Flux	≤ 0.5 second*
3.	Power Range, Neutron Flux, High Positive Rate	N.A.
4.	Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5.	Intermediate Range, Neutron Flux	N.A.
6.	Source Range, Neutron Flux	≤ 0.5 second*
7.	Overtemperature ΔT	≤ ~~4.0~~ 8.0 seconds**
8.	Overpower ΔT	~~N.A.~~ ≤ 8.0 seconds **
9.	Pressurizer Pressure-Low (Above P-7)	≤ 2.0 seconds
10.	Pressurizer Pressure-High	≤ 2.0 seconds
11.	Pressurizer Water Level-High (Above P-7)	N.A.
12.	Low Reactor Coolant Flow - Low	
	a. Single Loop (Above P-8)	≤ 1.0 second
	b. Two Loops (Above P-7 and below P-8)	≤ 1.0 second
13.	Steam Generator Water Level-Low-Low	≤ 2.0 seconds

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* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

** ~~Thermal lag and RTD bypass manifold delay times are not included.~~

Total time delay (including RTD time response and trip circuit channel electronic delays) from the time the temperature difference in the coolant loop exceeds the trip setpoint until the rods are free to fall (including time for trip break to open and CRDM gripper release).