ATTACHMENT A

PROPOSED CHANGES TO APPENDIX A, TECHNICAL SPECIFICATIONS,

OF FACILITY OPERATING LICENSES NPF-37 AND NPF-60

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TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤111.1% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤30.9% of RTP*
6. Source Range, Neutron Flux	17.0	10.0	0	≤10^5 cps	≤1.4×10^5 cps
7. Overtemperature ΔT	9.7 (27.7)#	5.85 (5.38)#	See Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.8 (4.3)#	1.22 (1.3)#	1.2	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	2.21	1.5	≥1885 psig	≥1871 psig
10. Pressurizer Pressure-High	3.1	0.71	1.5	≤2385 psig	≤2396 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span

*RTP = RATED THERMAL POWER

The value in parentheses only applies to Unit 2 until the RTD bypass manifolds are eliminated on this Unit.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	2.5	1.42 (1.77)#	0.6	≥90% of loop minimum measured flow*	≥ 88.8 (89.2)# % of loop minimum measured flow*
13. Steam Generator Water Level Low-Low					
a. Unit 1	27.1	18.28	1.5	≥40.8% of narrow range instrument span	≥39.1% of narrow range instrument span
b. Unit 2	17.0	14.78	1.5	≥17% of narrow range instrument span	≥15.3% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	12.0	0.7	0	≥5268 volts - each bus	≥4728 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	14.4	13.3	0	≥57.0 Hz	≥56.5 Hz
16. Turbine Trip					
a. Emergency Trip Header Pressure	N.A.	N.A.	N.A.	≥540 psig	≥520 psig
b. Turbine Throttle Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	N.A.	N.A.

*Minimum measured flow = 97,600 gpm

The value in parentheses only applies to Unit 2 until the RTD bypass manifolds are eliminated on this Unit.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

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,

ΔT_o = Indicated ΔT at RATED THERMAL POWER,

K_1 = 1.164,

K_2 = 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,

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

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s,

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

K_3 = 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 $-\infty$% and +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 +10%, the ΔT Trip Setpoint shall be automatically reduced by 2.0% 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 3.2 (3.9)# % of ΔT span.

The value in parentheses only applies to Unit 2 until the RTD bypass manifolds are eliminated on this Unit.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.00170/°F for T > T" and K_6 = 0 for T $\leq$ T",

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 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 2.7 (2.6)# % of ΔT span.

NOTE 5: The sensor error for temperature is 1.7 (1.2)# and for pressure is 1.1 (1.0)#.

The value in parentheses only applies to Unit 2 until the RTD bypass manifolds are eliminated on this Unit.

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. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

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.

thermal delays associated with thermowell-mounted temperature detectors (about 6 seconds) and thermal delays from the core to the loop temperature detectors and

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 3.3-2

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	N.A.
7. Overtemperature ΔT	≤~~4.0~~ 7.0 seconds*~~#~~
8. Overpower ΔT	N.A.
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.

*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.~~

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	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, 5)	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, 5)	Q	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5, 12)	Q(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R(13)#	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low (Above P-7)	S	R	Q**	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R	Q	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High (Above P-7)	S	R	Q	N.A.	N.A.	1

This note only applies to Unit 2 until the RTD bypass manifolds are eliminated on this Unit.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

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

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

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

(1) If not performed in previous 7 days.

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

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

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

(5) 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.

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

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

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

(9) Surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period.

(10) Setpoint verification is not applicable.

(11) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

(12) At least once per 18 months during shutdown verify that on a simulated Boron Dilution Doubling test signal CVCS valves 112D and E open and 112B and C close within 30 seconds.

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power					
a. ESF Bus Undervoltage	N.A.	N.A.	N.A.	2870 volts w/1.8s delay	≥2730 volts w/≤1.9s delay
b. Grid Degraded Voltage	N.A.	N.A.	N.A.	3804 volts w/310s delay	≥3728 volts w/310 ± 30s delay
9. Engineered Safety Feature Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤1930 psig	≤1936 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
c. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	≥550°F	≥ 547.0 (547.6)# °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.				

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure - Negative Rate - High is greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

This value only applies to Unit 2 until the RTD bypass manifolds are eliminated on this Unit.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided the Conditions a. through d. above are maintained. The combination of the RCS flow requirement (390,400 gpm) and the requirement on $F^N_{\Delta H}$ guarantee that the DNBR used in the safety analysis will be met.

A rod bow penalty is not applied to the final value of $F^N_{\Delta H}$ for the following reason:

Fuel rod bowing does reduce the value of the DNBR. However, predictions with the methods described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979 for the 17x17 Optimized Fuel Assemblies indicate that the fuel rod bow reduction on DNBR will be less than 3% at 33,000 MWD/MTU assembly average burnup. At higher burnups, the decrease in fissionable isotopes and the buildup of fission product inventory more than compensate for the rod bow reduction in DNBR.

There is a 11% margin available between the 1.32 and 1.34 design DNBR limits and the 1.47 and 1.49 safety analysis DNBR limit. Use of the 3% fuel rod bow DNBR margin reduction still leaves a 8% margin in DNBR between design limits and safety analysis limits.

The RCS flow requirement is based on the loop minimum measured flow rate of 97,600 gpm which is used in the Improved Thermal Design Procedure described in FSAR 4.4.1 and 15.0.3. A precision heat balance is performed once each cycle and is used to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% is assessed for potential feedwater venturi fouling. A maximum measurement uncertainty of 1.9 (2.2)# % has been included in the loop minimum measured flow rate to account for potential undetected feedwater venturi fouling and the use of the RCS flow indicators for flow rate verification. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken, before performing subsequent precision heat balance measurements, i.e., either the effect of fouling shall be quantified and compensated for in the RCS flow rate measurement, or the venturi shall be cleaned to eliminate the fouling.

Surveillance Requirement 4.2.3.4 provides adequate monitoring to detect possible flow reductions due to any rapid core crud buildup.

Surveillance Requirement 4.2.3.5 specifies that the measurement instrumentation shall be calibrated within seven days prior to the performance of the calorimetric flow measurement. This requirement is due to the fact that the drift effects of this instrumentation are not included in the flow measurement uncertainty analysis. This requirement does not apply for the instrumentation whose drift effects have been included in the uncertainty analysis.

The value in parentheses only applies to Unit 2 until the RTD bypass Manifolds are eliminated on this Unit.

<u>**ATTACHMENT B**</u>

<u>**LICENSING REPORT**</u>

WCAP-11323 Byron Units 1 & 2 and Braidwood Units 1 & 2 RTD Bypass Elimination Licensing Report (Proprietary)

WCAP-11324 Byron Units 1 & 2 and Braidwood Units 1 & 2 RTD Bypass Elimination Licensing Report (Non-Proprietary)

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

This proposed amendment involves removing and replacing the existing RTD bypass manifold system with fast response RTDs located in the reactor coolant hot leg and cold leg piping. The original RTD bypass system utilizes an arrangement which directs a sample of the RCS flow from the main coolant piping to an independent temperature measurement manifold. Coolant is redirected by scoops from the hot leg at three locations, 120° apart in the same plane around the pipe circumference, in order to obtain a representative sample. A cold leg sample is also taken, at the discharge of the reactor coolant pump, which provides sufficient mixing such that multiple sampling is not necessary. After temperature measurement, the sample is then returned to the main coolant flow. The narrow range RTDs provide the temperature to calculate loop delta-T and T_{ave}. In order to eliminate operating obstacles associated with the bypass system (such as leakage through valves, flanges, etc. and radiation exposure during maintenance) Commonwealth Edison is proposing to install a fast response system which measures loop temperature via RTDs protruding into the coolant flow within thermowells in the piping, thereby eliminating the bypass piping network.

The use of fast response RTDs does not involve a significant increase in the probability or consequences of any accident previously evaluated. Both the non-LOCA and LOCA accidents have been reviewed. Elimination of the RTD bypass system does not affect the probability of occurrence of the non-LOCA transients. The Uncontrolled RCCA Bank Withdrawal at Power and Loss of Load/Turbine Trip transients remain ANS Condition II events (moderate frequency) and the Steamline Rupture case analyzed remains ANS Condition IV (not expected to occur, limiting fault). The probability of occurrence of a LOCA (ANS Condition III, infrequent, for small break and Condition IV for large break) also remains unchanged as a result of this modification.

The non-LOCA and LOCA accidents were reviewed verifying that the variations in uncertainty associated with certain reactor trip functions, reflected in the technical specification changes, do not invalidate the current FSAR analyses of record and therefore, the design basis conclusions are still met. For non-LOCA transients, those that rely on overtemperature and overpower delta-T trips for mitigation were identified as limiting since the increased response time (an additional one second) would delay the generation of a trip signal slightly. These transients, which were specifically reanalyzed, were Uncontrolled RCCA Bank Withdrawal at Power, Loss of Load/Turbine Trip and Steamline Rupture at Power. Reanalysis of these transients confirmed that the FSAR conclusions remain valid and there is no increase in the consequences of any of these accidents. Additionally, it was determined that sufficient allowance exists in the current FSAR assumptions such that the total temperature measurement uncertainty for the new RTD system does not impact results. With respect to a LOCA, only nominal input values are used in the analysis and therefore, slight variations in uncertainties do not affect the results.

The use of fast response RTDs does not create the possibility of a new or different kind of accident from any accident previously evaluated. At Byron 1, the three hot leg RTDs and one cold leg RTD will utilize the existing penetrations into the RCS piping from the bypass system with only slight modifications. Therefore, the possibility of a new or different kind of accident does not exist. The only new penetration at this unit will be for the spare cold leg RTD. This thermowell boss will be designed, installed and inspected in accordance ASME Code Class 1 pressure boundary criteria. Caps and welds sealing the crossover leg bypass return piping nozzle, as well as the modification and welding for the existing penetrations, will be similarly qualified, thus precluding the possibility for a new or different kind of accident. At Byron 2, the three hot leg and two cold leg RTDs will be installed into new penetrations in the reactor coolant piping. The design, installation and inspection at this unit will also be qualified to ASME Code Class 1 criteria. By adherence to these industry standards, the pressure boundary integrity of the piping will be maintained, thus precluding the possibility for a new or different kind of accident.

Consideration has been given to plant response in the remote possibility that a thermowell would be ejected from its boss. It has been concluded that the affect of this flow area is insignificant on the results of the large break LOCA analyses and bounded by the results for small break LOCA analyses.

The function of the delta-T/T_{ave} protection channels is not changed because of the bypass elimination. The newly installed fast response RTDs perform the same function in both T_{hot} and T_{cold} applications. The three T_{hot} signals are averaged, with an allowance in the electronics to permit manually adding a bias to a two-RTD average should one RTD fail. A spare RTD is installed in the cold leg and can be manually activated should the other

RTD fail. These measured temperature values will still serve as input to two-out-of-four voting logic for protection functions. The basis for the instrumentation and control design meets the criteria of applicable IEEE standards, regulatory guides and general design criteria in that such principals as electrical separation, seismic and environmental qualification and single failure criteria are satisfied. Therefore, there is no possibility of a new or different kind of accident as a result of the instrumentation aspects of RTD bypass elimination.

Finally, the affect of the increased response time and setpoint uncertainty does not involve a significant reduction in a margin of safety. The investigation of the affect of these variables on non-LOCA and LOCA transients has verified that plant operation will be maintained within the bounds of safe, analyzed conditions as defined in the FSAR with the revised technical specifications. Conclusions presented in the FSAR remain valid even after taking into consideration the one second increase in response time of the new RTD system. As such, no reduction in the margin of safety between the FSAR acceptance limit and the ultimate safety limit (such as DNBR or pressure) has taken place for operation with the new RTD system.

For all the reasons stated above, Commonwealth Edison believes the proposed modification to the RTD bypass system does not involve any significant hazards considerations.

ATTACHMENT D

APPLICATION FOR WITHHOLDING

WESTINGHOUSE LETTER CAW-86-113

AFFIDAVIT OF ROBERT A. WIESEMANN

PROPRIETARY INFORMATION NOTICE

2529K