April 13, 1993

Docket Nos. STN 50-454, STN 50-455
and STN 50-456, STN 50-457

Mr. D. L. Farrar
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Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M83216, M83217, M83218, AND M83219)

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 53 to Facility Operating License No. NPF-37 and Amendment No. 53 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 42 to Facility Operating License No. NPF-72 and Amendment No. 42 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated April 15, 1992, as supplemented November 23, 1992.

The amendments revise the Technical Specifications to incorporate the results of the Commonwealth Edison Company's Setpoint Reconciliation Program.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original Signed By:

John B. Hickman, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 53 to NPF-37
2. Amendment No. 53 to NPF-66
3. Amendment No. 42 to NPF-72
4. Amendment No. 42 to NPF-77
5. Safety Evaluation

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See next page

*see previous page for concurrence

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. NPF-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated April 15, 1992, as supplemented November 23, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-37 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 53 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett for

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1993

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. NPF-66

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated April 15, 1992, as supplemented November 23, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 53 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett for

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1993

ATTACHMENT TO LICENSE AMENDMENT NOS. 53 AND 53

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND STN 50-455

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Pages indicated with an asterisk (*) are provided for convenience.

Remove Pages	Insert Pages
2-3	2-3
2-4	2-4
2-5	2-5
2-6	2-6
*2-7	*2-7
2-8	2-8
*2-9	*2-9
2-10	2-10
B 2-3	B 2-3
*B 2-4	*B 2-4
3/4 3-13	3/4 3-13
*3/4 3-14	*3/4 3-14
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2	B 3/4 3-2

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent within the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3-1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	≤109% of RTP*	≤111.36% of RTP*
b. Low Setpoint	≤25% of RTP*	≤27.36% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	≤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	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	≤25% of RTP*	≤31.5% of RTP*
6. Source Range, Neutron Flux	$\leq 10^5$ cps	$\leq 1.42 \times 10^5$ cps
7. Overtemperature ΔT	See Note 1	See Note 2
8. Overpower ΔT	See Note 3	See Note 4
9. Pressurizer Pressure-Low	≥1885 psig	≥1869 psig
10. Pressurizer Pressure-High	≤2385 psig	≤2393 psig
11. Pressurizer Water Level-High	≤92% of instrument span	≤93.5% of instrument span

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	≥90% of loop minimum measured flow*	≥89.3% of loop minimum measured flow*
13. Steam Generator Water Level Low-Low		
a. Unit 1	≥33.0% of narrow range instrument span	≥31.0% of narrow range instrument span
b. Unit 2	≥36.3% of narrow range instrument span	≥34.8% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	≥5268 volts - each bus	≥4920 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	≥57.0 Hz	≥56.8 Hz
16. Turbine Trip		
a. Emergency Trip Header Pressure	≥1000 psig	≥815 psig
b. Turbine Throttle Valve Closure	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.

*Minimum measured flow = 97,600 gpm

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
19. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7		
1) P-10 input	$\leq$10% of RTP*	$\geq$7.9% to $\leq$12.1% of RTP*
2) P-13 input	<10% RTP* Turbine Impulse Pressure Equivalent	<12.1% RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	$\leq$30% of RTP*	$\leq$32.1% of RTP*
d. Power Range Neutron Flux, P-10	$\leq$10% of RTP*	$\geq$7.9% to $\leq$12.1% of RTP*
e. Turbine Impulse Chamber Pressure, P-13	<10% RTP* Turbine Impulse Pressure Equivalent	<12.1% RTP* Turbine Impulse Pressure Equivalent
20. Reactor Trip Breakers	N.A.	N.A.
21. Automatic Trip and Interlock Logic	N.A.	N.A.
22. Reactor Trip Bypass Breakers	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + t_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

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, $\tau_1 = 8$ s, $\tau_2 = 3$ s,
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT,
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ 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}, $\tau_4 = 33$ s, $\tau_5 = 4$ 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^1 $\leq$ 588.4°F (Nominal T_{avg} at RATED THERMAL POWER),

K_3 = 0.00134,

P = Pressurizer pressure, psig,

P^1 = 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% and +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 +13%, the ΔT Trip Setpoint shall be automatically reduced by 1.74% of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds -32%, the ΔT Trip Setpoint shall be automatically reduced by 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 3.71% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

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_0	=	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	=	$0.00170/^{\circ}F$ 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, $\leq 588.4^{\circ}F$),
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% of ΔT span.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

The methodology to derive the Trip Setpoints is based upon combing all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore, providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the accident analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity addition that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Cooling Fans, Control Room Isolation, Phase "A" Isolation, Turbine Trip, Auxiliary Feedwater, Containment Vent Isolation and Essential Service Water)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-1	≤ 3.4 psig	≤ 4.6 psig
d. Pressurizer Pressure-Low (Above P-11)	≥ 1829 psig	≥ 1813 psig
e. Steam Line Pressure-Low (Above P-11)	≥ 640 psig*	≥ 614 psig*
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-3	≤ 20.0 psig	≤ 21.2 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
b. Phase "B" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure-High-3	$\leq$ 20.0 psig	$\leq$ 21.2 psig
c. Containment Vent Isolation		
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
2) Manual Phase "A" Isolation	N.A.	N.A.
3) Manual Phase "B" Isolation	N.A.	N.A.
4) Safety Injection	See Item 1 above for all Safety Injection Trip Setpoints and Allowable Values.	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)		
f. Division 11 for Unit 1 (Division 21 for Unit 2) ESF Bus Undervoltage-Start Motor-Driven Pump	2870 volts	2730 volts
g. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to Essential Service Water)	1.22" Hg vac	2" Hg vac
7. Automatic Opening of Containment Sump Suction Isolation Valves		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. RWST Level-Low-Low Coincident with Safety Injection	46.7%	44.7%

See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.

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
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.	

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

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

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

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statisitical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, and (11) essential service water pumps start and automatic valves position.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
License No. NPF-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated April 15, 1992, as supplemented November 23, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 42 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett for

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1993

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
License No. NPF-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated April 15, 1992, as supplemented November 23, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 42 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date if its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett for

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1993

ATTACHMENT TO LICENSE AMENDMENT NOS. 42 AND 42

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. Pages indicated with an asterisk (*) are provided for convenience.

Remove Pages	Insert Pages
2-3	2-3
2-4	2-4
2-5	2-5
2-6	2-6
*2-7	*2-7
2-8	2-8
*2-9	*2-9
2-10	2-10
B 2-3	B 2-3
*B 2-4	*B 2-4
3/4 3-13	3/4 3-13
*3/4 3-14	*3/4 3-14
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-26a (Unit 2 only)	--
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2	B 3/4 3-2

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent within the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	≤109% of RTP*	≤111.36% of RTP*
b. Low Setpoint	≤25% of RTP*	≤27.36% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	≤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	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	≤25% of RTP*	≤31.5% of RTP*
6. Source Range, Neutron Flux	$\leq 10^5$ cps	$\leq 1.42 \times 10^5$ cps
7. Overtemperature ΔT	See Note 1	See Note 2
8. Overpower ΔT	See Note 3	See Note 4
9. Pressurizer Pressure-Low	≥1885 psig	≥1869 psig
10. Pressurizer Pressure-High	≤2385 psig	≤2393 psig
11. Pressurizer Water Level-High	≤92% of instrument span	≤93.5% of instrument span

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

*Minimum measured flow = 97,600 gpm

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
19. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7		
1) P-10 input	$\leq$10% of RTP*	$\geq$7.9% to $\leq$12.1% of RTP*
2) P-13 input	<10% RTP* Turbine Impulse Pressure Equivalent	<12.1% RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	$\leq$30% of RTP*	$\leq$32.1% of RTP*
d. Power Range Neutron Flux, P-10	$\leq$10% of RTP*	$\geq$7.9% to $\leq$12.1% of RTP*
e. Turbine Impulse Chamber Pressure, P-13	<10% RTP* Turbine Impulse Pressure Equivalent	<12.1% RTP* Turbine Impulse Pressure Equivalent
20. Reactor Trip Breakers	N.A.	N.A.
21. Automatic Trip and Interlock Logic	N.A.	N.A.
22. Reactor Trip Bypass Breakers	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

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_0	=	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 -32% and +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 13% the ΔT Trip Setpoint shall be automatically reduced by 1.74% of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds -32%, the ΔT trip setpoint shall be automatically reduced by 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 3.71% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \frac{(\tau_7 S)}{1 + \tau_7 S} \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T\left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

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_0	=	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	=	$0.00170/^{\circ}F$ 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, $\leq 588.4^{\circ}F$),
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% of ΔT span.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore, providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the accident analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity addition that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Cooling Fans, Control Room Isolation, Phase "A" Isolation, Turbine Trip, Auxiliary Feedwater, Containment Vent Isolation and Essential Service Water)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-1	≤ 3.4 psig	≤ 4.6 psig
d. Pressurizer Pressure-Low (Above P-11)	≥ 1829 psig	≥ 1813 psig
e. Steam Line Pressure-Low (Above P-11)	≥ 640 psig*	≥ 614 psig*
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-3	≤ 20.0 psig	≤ 21.2 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
b. Phase "B" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure-High-3	≤ 20.0 psig	≤ 21.2 psig
c. Containment Vent Isolation		
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
2) Manual Phase "A" Isolation	N.A.	N.A.
3) Manual Phase "B" Isolation	N.A.	N.A.
4) Safety Injection	See Item 1 above for all Safety Injection Trip Setpoints and Allowable Values.	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)		
f. Division 11 for Unit 1 (Division 21 for Unit 2) ESF Bus Undervoltage-Start Motor-Driven Pump	2870 volts	2730 volts
g. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to Essential Service Water)	1.22" Hg vac	2" Hg vac
7. Automatic Opening of Containment Sump Suction Isolation Valves		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. RWST Level-Low-Low Coincident with Safety Injection	46.7%	44.7%
See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.		

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
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.	

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

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

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

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statisitical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, and (11) essential service water pumps start and automatic valves position.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

1.0 INTRODUCTION

In a letter dated April 15, 1992, as supplemented November 23, 1992, Commonwealth Edison Company (CECo, the licensee) proposed changes to the Technical Specifications (TS) for both Byron and Braidwood Stations, Units 1 and 2. The proposed TS changes are the result of a setpoint reconciliation program initiated by CECo for the Byron and Braidwood stations. The specifics of the setpoint study were previously presented to the NRC by CECo at their Chicago, Illinois office on April 30, 1991, and again on August 6, 1991, where program results were presented and a follow-up action plan was proposed. The setpoint reconciliation program resulted in proposed revisions to TS Section 2.2-1, Table 2.2-1, TS Section 3.3.2, Table 3.3-4 and the associated Bases. The licensee stated that 25 setpoints were to be evaluated in the setpoint study. The November 23, 1992, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

In November 1990, Westinghouse issued a revision of the Byron/Braidwood setpoint bases (error analysis) document. Commonwealth Edison Company engineering noted inconsistencies with the Measurement and Test Equipment accuracy (MTE) and other setpoint error assumptions incorporated by Westinghouse. Commonwealth Edison Company engineering evaluated the revised Westinghouse bases document and determined that some values listed in the Byron/Braidwood TS could be affected (Total Allowance, Z, Sensor Error, Trip Setpoint, and Allowable Value). Subsequent meetings were held with the NRC where CECo committed to an evaluation of the Westinghouse assumptions and revised bases document to confirm the continued Westinghouse setpoint bases applicability to Byron/Braidwood stations.

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3.0 EVALUATION

The Byron/Braidwood TS Section 2.2.1, Table 2.2-1, TS Section 3.3-2 and Table 3.3-4 are of the five column type listing values for Total Allowance (TA), Z, Sensor Error (SE), Trip Setpoint, and Allowable Value (AV). The inclusion of values of Z, SE and TA provide a means within the Westinghouse setpoint methodology to determine channel operability should the "as measured" rack or sensor allowances exceed the TS allowable value (trigger value). The initial setpoint values for the reactor trip system (RTS) were developed by Westinghouse and the Engineered Safety Features Actuation System (ESFAS) values were determined by Sargent and Lundy/Westinghouse.

The licensee found inconsistencies with the assumptions used by Westinghouse in the setpoint bases document. Among these discrepancies were the error terms for MTE ratio that considered better accuracy than actual plant measurement and test equipment being used for calibration. In addition the as-built condition (instrument type, options) did not always agree with the Westinghouse assumptions. The licensee confirmed that 25 setpoints were evaluated in the setpoint study. Of these setpoints, 10 initially resulted in zero or negative margin with respect to the safety analysis (CSA+Margin was inadequate). By improving the accuracy of the error terms and calibration process the licensee was able to demonstrate positive margins for 8 of the 10 setpoints. The two remaining setpoints, Steam Generator Level Hi-Hi (Model D4) and reactor coolant system (RCS) Low Flow required re-evaluation of the safety analysis by Westinghouse to include the revised Byron/Braidwood plant specific error terms. The results obtained by Westinghouse showed that there was sufficient analytical margin to allow a revision to the safety analysis limit without affecting the result of the safety analysis. Commonwealth Edison Company engineering concluded that a TS revision to Tables 2.2-1 and 3.3-4 for TA, Z, SE, and Allowable value were required.

Based on the results of the setpoint study the revised values for TA, SE, and Z, in many cases, would no longer satisfy the TS equation 2.2-1, $Z+RE+SE \leq TA$. Equation 2.2-1 provides a licensee a method to determine instrumentation operability when the "as measured" rack drift exceeds the allowable value and provides a threshold value for reportable events. The equation is no longer satisfied based in part on the licensee's decision to use an available or developed margin to maintain instrument setpoints by revising the allowable values. The Westinghouse methodology states that with reduced margin, the use of SE in Table 2.2-1 will result in the sum of Z+RE+SE being greater than TA. The licensee has, therefore, proposed that the values for SE, RE and TA be removed from the TS. The TS 2.2-1 equation $Z+RE+SE \leq TA$ will also be deleted. The deletion of values for Z, SE, and TA from the TS limits a licensee's flexibility in responding to instrumentation channels found outside the allowable value. However, an allowable value is acceptable in determining instrumentation operability and is based on industry standards and practice. The licensee states the calibration tolerance controls the setpoint within procedures and is the normal basis for determining when to recalibrate and confirm instrument operability. The licensees setpoint methodology is based on the methodology developed by Westinghouse and presented in WCAP-12583,

"Westinghouse Setpoint Methodology for Protection Systems, May 1990." The resulting three column TS format is acceptable to the staff.

An additional change requested by the licensee is a revision to Functional Unit 16.a, Turbine Trip, Emergency Trip Header Pressure. The proposed change is the result of an installation of a new pressure switch designed for higher fluid pressure and not a direct result of the CECo setpoint study. The resulting trip setpoint and allowable value are greater than the values currently specified in the TS. The setpoint calculations for this functional unit were not part of the CECo setpoint study but were developed by Sargent and Lundy. The licensee stated that the revised setpoint will result in a anticipatory reactor trip earlier than that provided by the present setpoint. This is acceptable to the staff.

The licensee's proposed TS amendment requires equipment changes and revised test equipment requirements be implemented prior to adopting the CECo setpoint study results. The setpoint MTE assumptions require the use of new measurement and test equipment. The installation of temperature compensation transmitters is also assumed by the study. In addition, more stringent calibration tolerances are required for some bistables. The above calibration and equipment modifications are intended to ensure that the revised setpoint study reflects plant as-built and calibration procedures/practices.

4.0 PROPOSED CECo SETPOINT STUDY TS CHANGES

4.1 BYRON UNITS 1 AND 2

2.2 LIMITING SAFETY SYSTEM SETTINGS
REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

1. TS 2.2.1, ACTION b.1 is deleted and ACTION b.2 is merged with ACTION b.
2. Equation 2.2-1 is deleted along with definitions for Z, RE, SE, and TA.

TABLE 2.2-1 REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

1. The columns for Total Allowance (TA), Z, and Sensor Error (SE) have been deleted for all Functional Units of Table 2.2-1.

2. The Allowable Values for the following Functional Units have been revised as a result of the CECo setpoint study:

   2. Power Range, Neutron flux

      a. High Setpoint
      b. Low Setpoint

   5. Intermediate Range, Neutron Flux

   6. Source Range, Neutron Flux

9. Pressurizer Pressure - Low

10. Pressurizer Pressure - High

11. Pressurizer Water Level - High

12. Reactor Coolant Flow - Low

13. Steam Generator Water Level Low - Low

    b. Unit 2

14. Undervoltage - Reactor Coolant Pumps

15. Underfrequency - Reactor Coolant Pumps

16. Turbine Trip

    a. Emergency Trip Header Pressure

3. The setpoint for Functional Unit 16.a, Table 2.2-1, Turbine Trip, Emergency Trip Header Pressure was revised to reflect a new instrument installation.

4. Note 1, Parts (i), (ii), and (iii) have been revised to delete information specific to previous fuel cycles.

5. Table 2.2-1, Note 2; revised the value that the maximum trip setpoint should not exceed the computed trip setpoint value.

6. Table 2.2-1, Note 4; revised the value that the maximum trip setpoint should not exceed the computed setpoint value.

7. Note 5 has been deleted. The values for sensor errors are no longer applicable.

BASES
2.2-1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The optional provision to determine the operability of a channel when the trip setpoint is found to exceed the allowable value has been deleted.

INSTRUMENTATION
3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
LIMITING CONDITION FOR OPERATION

1. TS 3.3.2, ACTION b.1 is deleted and ACTION b.2 is merged with Action b.
2. Equation 2.2-1 is deleted along with definitions for Z, RE, SE, and TA.

TABLE 3.3-4 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SETPOINTS

1. The columns for Total Allowance (TA), Z, and Sensor Error (SE) have been deleted for all Functional Units of Table 3.3-4.

2. The Allowable Value has been revised based on the CECo setpoint study results for the following functional units:

   1. Safety Injection

      c. Containment Pressure - High-1
      d. Pressurizer Pressure - Low (Above P-11)
      e. Steam Line Pressure - Low (Above P-11)

   2. Containment Spray

      c. Containment Pressure - High-3

   3. Containment Isolation

      b.3 Containment Pressure - High-3

   4. Steam Line Isolation

      c. Containment Pressure - High-2
      d. Steam Line Pressure - Low (Above P-11)
      e. Steam Line Pressure Negative Rate-High (Below P-11)

   5. Turbine Trip and Feedwater Isolation

      b. Steam Generator Water Level High-High (P-14)

         1) Unit 1

   6. Auxiliary Feedwater

      c. Steam Generator Water Level - Low-Low-Start Motor-Driven Pump and Diesel-Driven Pump

         2) Unit 2

      d. Undervoltage-RCP Bus-Start Motor-Driven Pump and Diesel-Driven Pump

   9. Engineered Safety Feature Actuation System Interlocks

      c. Low-Low T_{avg}, P-12

3/4.3 INSTRUMENTATION
BASES
3/4.3.1 AND 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The optional provision to determine the operability of a channel when the trip setpoint is found to exceed the allowable value has been deleted.

4.2 BRAIDWOOD UNITS 1 AND 2

2.2 LIMITING SAFETY SYSTEM SETTINGS
REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

1. TS 2.2.1, ACTION b.1 is deleted and ACTION b.2 is merged with ACTION b.
2. Equation 2.2-1 is deleted along with definitions for Z, RE, SE, and TA.

TABLE 2.2-1 REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

1. The columns for Total Allowance (TA), Z, and Sensor Error (SE) have been deleted for all Functional Units of Table 2.2-1.

2. Note 1, Parts (i), (ii), and (iii) have been revised to delete information specific to previous fuel cycles.

3. Note 5 has been deleted. The values for sensor errors are no longer applicable.

4. The Allowable Values for the following functional units have been revised as a result of the CECo setpoint study:

   2. Power Range, Neutron flux

      a. High Setpoint
      b. Low Setpoint

   5. Intermediate Range, Neutron Flux

   6. Source Range, Neutron Flux

   9. Pressurizer Pressure - Low

   10. Pressurizer Pressure - High

   11. Pressurizer Water Level - High

   12. Reactor Coolant Flow - Low

   13. Steam Generator Water Level Low - Low

      b. Unit 2

14. Undervoltage - Reactor Coolant Pumps

15. Underfrequency - Reactor Coolant Pumps

16. Turbine Trip

    a. Emergency Trip Header Pressure

5. Table 2.2-1, Note 2; revised the value that the maximum trip setpoint should not exceed the computed trip setpoint.

6. Table 2.2-1, Note 4; revised the value that the maximum trip setpoint should not exceed the computed setpoint.

7. The setpoint for Functional Unit 16a, Table 2.2-1, Turbine Trip, Emergency Trip Header Pressure was revised to reflect a new instrument installation.

BASES
2.2-1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The optional provision to determine the operability of a channel when the trip setpoint is found to exceed the allowable value has been deleted.

INSTRUMENTATION
3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
LIMITING CONDITION FOR OPERATION

1. TS 3.3.2, ACTION b.1 is deleted and ACTION b.2 is merged with Action b.
2. Equation 2.2-1 is deleted along with definitions for Z, RE, SE, and TA.
3. Removed footnote *. Applicable to Cycle 1 only.

TABLE 3.3-4 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

1. The columns for Total Allowance (TA), Z, and Sensor Error (SE) have been deleted for all Functional Units of Table 3.3-4.

2. The Allowable Value has been revised based on the CECo setpoint study results for the following functional units:

    1. Safety Injection

        c. Containment Pressure - High-1
        d. Pressurizer Pressure - Low (Above P-11)
        e. Steam Line Pressure - Low (Above P-11)

    2. Containment Spray

        c. Containment Pressure - High-3

3. Containment Isolation

   b.3 Containment Pressure - High-3

4. Steam Line Isolation

   c. Containment Pressure - High-2
   d. Steam Line Pressure - Low (Above P-11)
   e. Steam Line Pressure Negative Rate-High (Below P-11)

5. Turbine Trip and Feedwater Isolation

   b. Steam Generator Water Level High-High (P-14)

      1) Unit 1
      2) Unit 2

6. Auxiliary Feedwater

   c. Steam Generator Water Level-Low-Low-Start Motor-Driven Pump and Diesel-Driven Pump

      2) Unit 2

   d. Undervoltage-RCP Bus-Start Motor-Driven Pump and Diesel-Driven Pump

9. Engineered Safety Feature Actuation System Interlocks

   c. Low-Low T_{avg}, P-12

3/4.3 INSTRUMENTATION
BASES
3/4.3.1 AND 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The optional provision to determine the operability of a channel when the trip setpoint is found to exceed the allowable value has been deleted.

5.0 SUMMARY

Based on the above evaluation, the staff finds the proposed changes to the Byron and Braidwood TS to be acceptable. The revised allowable values and setpoints were developed using an accepted setpoint methodology. The CECo setpoint study incorporates plant specific error terms and assumptions and is, therefore, representative of plant as-built conditions and surveillance requirements. However, the CECo setpoint study results and resultant TS revisions are dependent on the licensee's use of new test and measurement equipment, temperature compensation transmitters, and revised calibration procedures. The licensee's decision to revise the Byron and Braidwood TS to a

three column format and delete the values of TA, SE and Z from Tables 2.2-1 and 3.3-4 based on the CECo setpoint study results, is also acceptable to the staff.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (57 FR 24666). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: C. Doutt

Date: April 13, 1993