UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
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 February 21, 1995, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 74 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

for

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 5, 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
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 February 21, 1995, 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. 74 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

for

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 5, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 74 AND 74

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 only.

Remove Pages	Insert Pages
2-7	2-7
2-8	2-8
2-10	2-10
B 2-5	B 2-5
*B 2-6	*B 2-6
3/4 3-12a	3/4 3-12a
3/4 3-28	3/4 3-28

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT	=	Measured ΔT by RTD 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,# ≤ 2s,##
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER,
K_1	=	1.325*, (1.164)**
K_2	=	0.0297/°F*, (0.0265/°F)**
$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation,
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 33 s, τ_5 = 4 s,
T	=	Average temperature, °F,

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

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

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

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

T′ ≤ 588.4°F (Nominal T_{avg} at RATED THERMAL POWER),

K_3 = 0.00181[*], (0.00134)[**]

P = Pressurizer pressure, psig,

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

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

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

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

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

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

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

[*]Applicable to Unit 1. Applicable to Unit 2 after cycle 5.
[**]Not applicable to Unit 1. Applicable to Unit 2 until completion of cycle 5.
[#]Applicable to Unit 1 until the completion of cycle 7. Applicable to Unit 2 until the completion of cycle 6.
[##]Applicable to Unit 1 starting with cycle 8. Applicable to Unit 2 starting with cycle 7.

NOTE 3: (Continued)

K_6 = 0.00245/°F* (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, ≤ 588.4°F),

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

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

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

Power Range, Neutron Flux, High Rates (Continued)

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

Intermediate and Source Range, Neutron Flux

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

Overtemperature ΔT

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

Overpower ΔT

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

Pressurizer Pressure

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

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

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

Pressurizer Water Level

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

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(10) Setpoint verification is not applicable.

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

(12) Not used.

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

(14) Verify that the appropriate signals reach the Undervoltage and Shunt Trip Relays, for both the Reactor Trip and Bypass Breakers from the Manual Trip Switches.

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

(16) Automatic Undervoltage trip.

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 66
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 February 21, 1995, 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. 66 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

Ramin R. Assa, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 5, 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. [illegible]
License No. [illegible]-77

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated February 21, 1995, 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. 66 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 licensc. 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

Ramin R. Assa, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 5, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 66 AND 66

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 only.

Remove Pages	Insert Pages
2-7	2-7
2-8	2-8
2-10	2-10
B 2-5	B 2-5
*B 2-6	*B 2-6
3/4 3-12a	3/4 3-12a
3/4 3-28	3/4 3-28

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:	ΔT	=	Measured ΔT by RTD 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*, ≤2s,**
	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER,
	K_1	=	1.164,* 1.325**
	K_2	=	0.0265/°F,* 0.0297/°F**
	$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation,
	τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 33 s, τ_5 = 4 s,
	T	=	Average temperature, °F,

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

NOTE 1: (Continued)

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

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

T′ ≤ 588.4°F (Nominal T_{avg} at RATED THERMAL POWER),

K_3 = 0.00134*, 0.00181**

P = Pressurizer pressure, psig,

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

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

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

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

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

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

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

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

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00170/°F*, 0.00245/°F** for T > T" and K_6 = 0 for T ≤ T",
T	=	As defined in Note 1,
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 588.4°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

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

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

Power Range, Neutron Flux, High Rates (Continued)

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

Intermediate and Source Range, Neutron Flux

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

Overtemperature ΔT

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

Overpower ΔT

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

Pressurizer Pressure

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

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

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

Pressurizer Water Level

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

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(10) Setpoint verification is not applicable.

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

(12) Not Used.

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

(14) Verify that the appropriate signals reach the Undervoltage and Shunt Trip relays, for both the Reactor Trip and Bypass Breakers from the Manual Trip Switches.

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

(16) Automatic undervoltage trip.

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

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

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

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