Docket file

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

October 21, 1994

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M90575, M90576, M90577, AND M90578)

Dear Mr. Farrar:

The Commission has issued the enclosed Amendment No. 65 to Facility Operating License No. NPF-37 and Amendment No. 65 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 56 to Facility Operating License No. NPF-72 and Amendment No. 55 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 March 23, 1994, as supplemented on July 26, 1994.

The amendments change the Technical Specifications to reflect a reduced thermal flow to compensate for increased steam generator tube plugging up to 15 percent of the total number of tubes. The amendment also approves the use of higher boron concentration in the refueling water storage tank, the reactor coolant system accumulators, and the refueling cavity.

The March 23, 1994, submittal also requested approval of a positive moderator temperature coefficient. That portion of the request remains under staff review.

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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:

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

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

Enclosures:
1. Amendment No. 65 to NPF-37
2. Amendment No. 65 to NPF-66
3. Amendment No. 56 to NPF-72
4. Amendment No. 55 to NPF-77
5. Safety Evaluation

cc w/encl: see next page

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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. 65
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 March 23, 1994, as supplemented on July 26, 1994, 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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P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 65 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

George F. Dick, Jr.

George F. Dick, Jr., 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: October 21, 1994

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. 65
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 March 23, 1994, as supplemented on July 26, 1994, 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. 65 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

George F. Dick, Jr.

George F. Dick, Jr., 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: October 21, 1994

ATTACHMENT TO LICENSE AMENDMENT NOS. 65 AND 65

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 by an asterisk are provided for convenience only.

Remove Pages	Insert Pages
III	III
2-1	2-1
2-2	2-2
-	2-2a
2-5	2-5
2-7	2-7
2-8	2-8
2-10	2-10
B 2-1	B 2-1
3/4 1-11	3/4 1-11
3/4 1-12	3/4 1-12
3/4 2-8	3/4 2-8
3/4 5-1	3/4 5-1
3/4 5-11	3/4 5-11
3/4 9-1	3/4 9-1
3/4 9-2*	3/4 9-2*
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 2-3*	B 3/4 2-3*
B 3/4 2-4	B 3/4 2-4
B 3/4 5-4	B 3/4 5-4
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4*	B 3/4 6-4*
B 3/4 9-1	B 3/4 9-1

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2.1 SAFETY LIMITS

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2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS..... B 2-3

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1* (Figure 2.1-1a)** for four loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within this limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within this limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

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

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

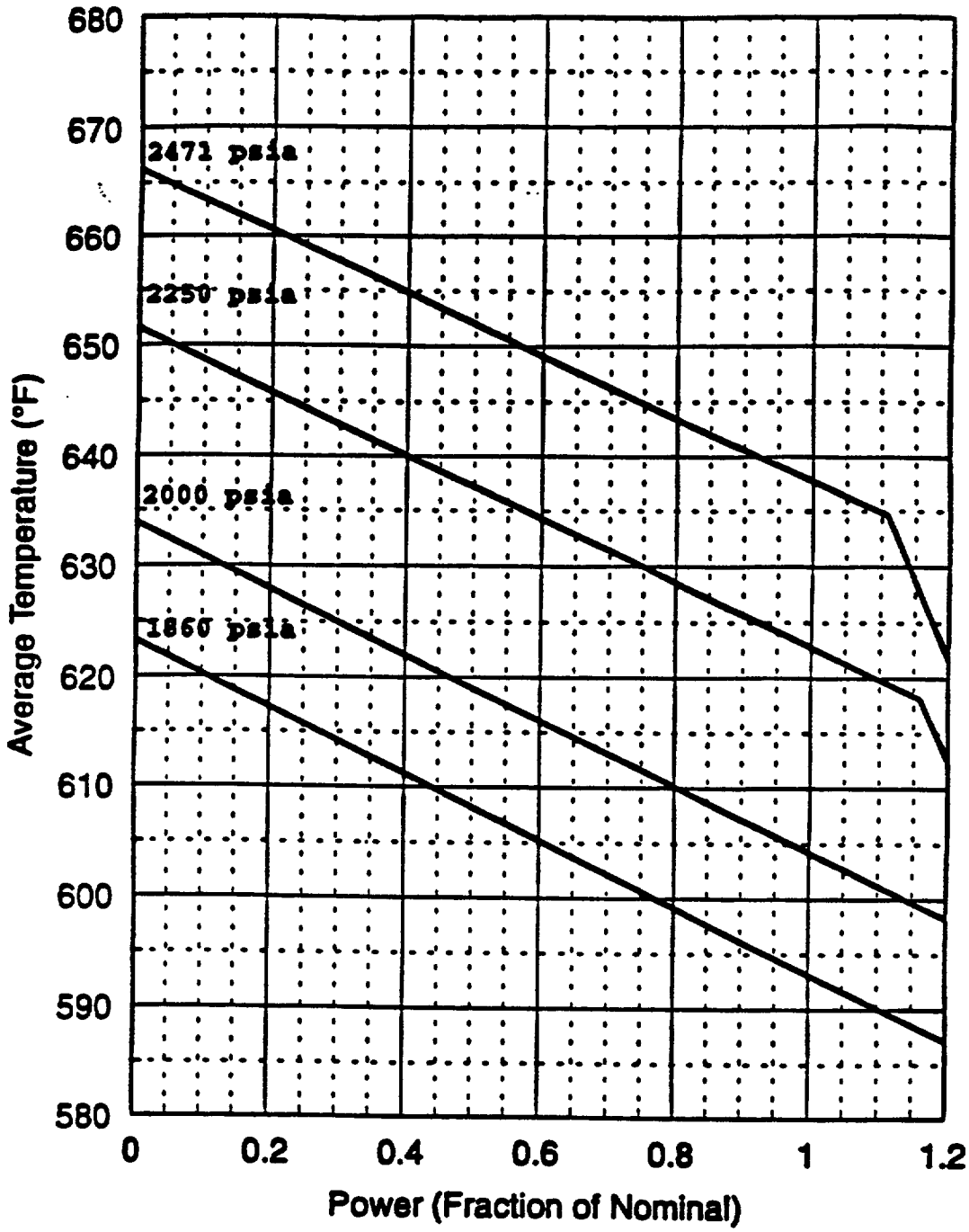

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

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

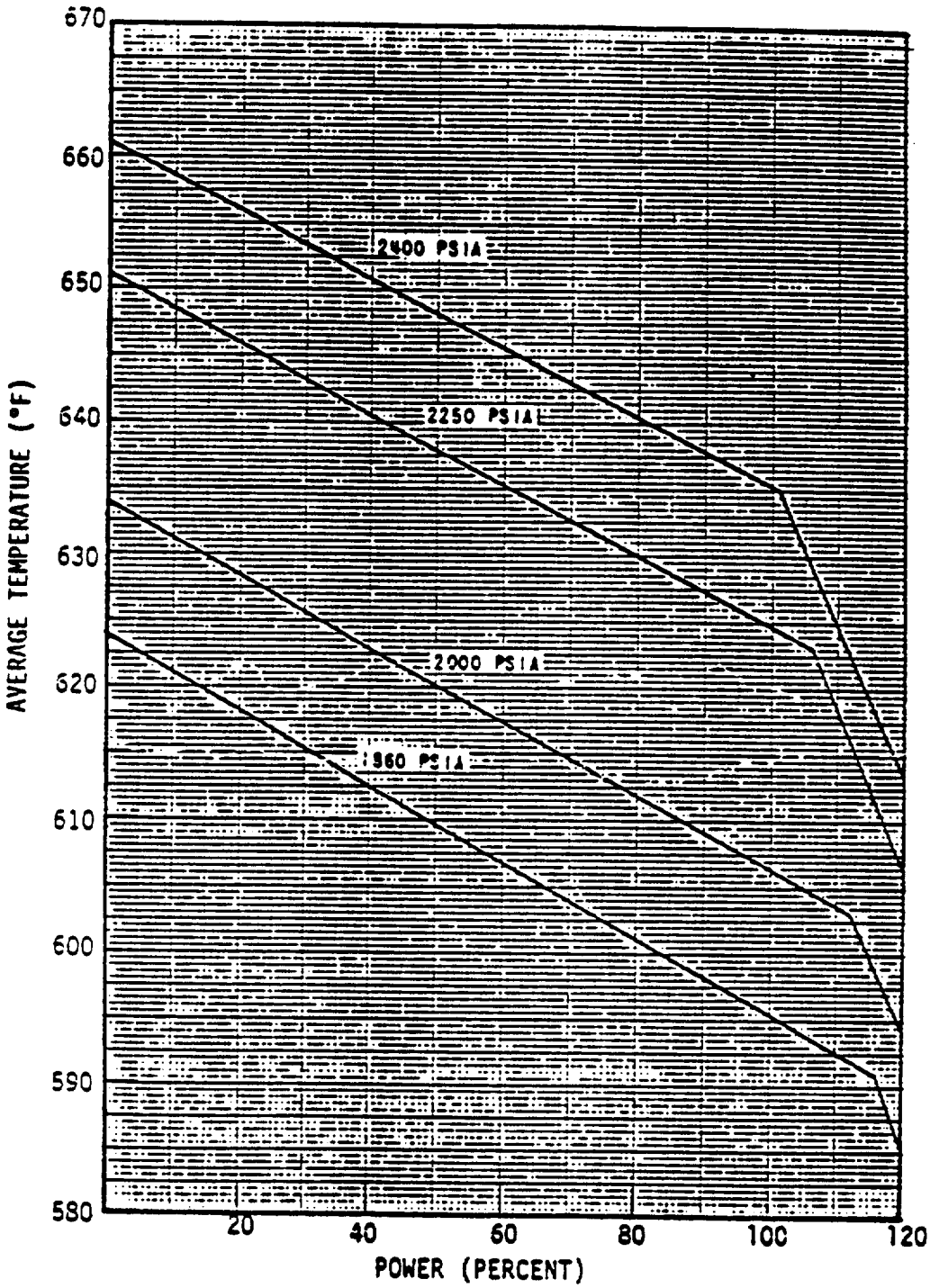

FIGURE 2.1-1a

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINTS	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	≥90% of loop minimum measured flow*	≥89.3% of loop minimum measured flow*
13. Steam Generator Water Level Low-Low		
a. Unit 1	≥33.0% of narrow range instrument span	≥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	≥56.08 Hz
16. Turbine Trip		
a. Emergency Trip Header Pressure	≥1000 psig	≥815 psig
b. Turbine Throttle Valve Closure	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.

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

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation,
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead-lag compensator on measured ΔT,
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s,
$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT,
τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3 = 0 s,
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER,
K_1	=	1.325*, (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.

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,

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%)** 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.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

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

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

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNBR thermal design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95% (at a 95% confidence level) for any Condition I or II event.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered. As described in the UFSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

The design DNBR values are 1.25 for the typical and thimble cells (1.34 and 1.32 for a typical cell and a thimble cell, respectively for OFA** fuel, and 1.33 for a typical cell and 1.32 for a thimble cell for the VANTAGE 5 fuel*). In addition, margin has been maintained in both designs by meeting safety analysis DNBR limits of 1.50 for the typical and thimble cells (1.49 for a typical cell and 1.47 for a thimble cell for OFA fuel, and 1.67 and 1.65 for a typical cell and a thimble cell, respectively for the VANTAGE 5 fuel*) in performing safety analyses.

The curves of Figure 2.1-1 (Figure 2.1-1a*) show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum design DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

---

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

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water level of 7.0%,

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water level of 9.0%,

2) a)* A boron concentration between 2300 and 2500 ppm,

b)** A minimum boron concentration of 2000 ppm, and

3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water level, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 35°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.

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2 for MODES 1, 2 and 3 and one of the following borated water sources shall be OPERABLE as required by Specification 3.1.2.1 for MODE 4:

a. A Boric Acid Storage System with:

1) A minimum contained borated water level of 40%,

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water level of 89%,

2) a)* A boron concentration between 2300 and 2500 ppm,

b)** A minimum boron concentration of 2000 ppm,

3) A minimum solution temperature of 35°F, and

4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources in MODE 1, 2, or 3, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable in MODE 1, 2, or 3, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With no borated water source OPERABLE in MODE 4, restore one borated water source to OPERABLE status within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

---

*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.

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 Indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows for four loop operation.

a. 1)* RCS Total Flowrate ≥371,400 gpm,

2)** RCS Total Flowrate ≥390,400 gpm, and

b. $F^N_{\Delta H} \leq 1.55\ [1.0 + 0.3\ (1.0-P)]$ for OFA fuel

$F^N_{\Delta H} \leq 1.65\ [1.0 + 0.3\ (1.0-P)]$ for VANTAGE 5 fuel

where:

Measured values of $F^N_{\Delta H}$ are obtained by using the movable incore detectors. An appropriate uncertainty of 4% (nominal) or greater shall then be applied to the measured value of $F^N_{\Delta H}$ before it is compared to the requirements, and

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore RCS total flow rate and $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux-High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

---

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water level of between 31% and 63%,

c. 1)# A boron concentration between 2200 and 2400 ppm,

2)** A boron concentration between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between 602 and 647 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water level and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

---

*Pressurizer pressure above 1000 psig.
#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.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) and the heat traced portion of the RWST vent path shall be OPERABLE with:

a. A minimum contained borated water level of 89%,

b. 1)* A boron concentration between 2300 and 2500 ppm,

   2)** A minimum boron concentration of 2000 ppm,

c. A minimum water temperature of 35°F, and

d. A maximum water temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1) Verifying the contained borated water level in the tank, and

   2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 35°F or greater than 100°F, and

c. At least once per 24 hours by verifying the RWST vent path temperature to be greater than or equal to 35°F when the outside air temperature is less than 35°F.

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

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. A K_{eff} of 0.95 or less, or

b. 1)# A boron concentration of greater than or equal to 2000 ppm.

2)** A boron concentration of greater than or equal to 2300 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm# (2300 ppm**), whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 57 steps (approximately 3 feet) from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 Valves CV111B, CV8428, CV8441, CV8435, and CV8439 shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days.

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*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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

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

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} $\Delta k/k/°F$. The MTC value of -3.2×10^{-4} $\Delta k/k/°F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.1×10^{-4} $\Delta k/k/°F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC can be maintained within its limits. The BOL MTC measurement, combined with the predicted MTC throughout core life, will be used to impose administrative limits on rod withdrawal, as required during core life to ensure that MTC will always be less positive than 0 $\Delta k/k/°F$. This coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 550°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor vessel is above its minimum RT_{NDT} temperature, and (5) the plant is above the cooldown steam dump permissive, P-12.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement is 13,487 (15,780*) gallons of 7000-ppm borated water from the boric acid storage tanks or 54,014 (70,450*) gallons of 2300-ppm (2000-ppm*) borated water from the refueling water storage tank. A Boric Acid Storage System level of 40% ensures that there is a volume of greater than or equal to 13,487 (15,780*) gallons available. A RWST level of 89% ensures that there is a volume of greater than or equal to 395,000 gallons available.

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

BORATION SYSTEMS (Continued)

With the RCS temperature below 350°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 330°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or an RHR Suction relief valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 740 (2,652*) gallons of 7000-ppm borated water from the boric acid storage tanks or 2264 (11,840*) gallons of 2300-ppm (2000-ppm*) borated water from the refueling water storage tank (RWST). A Boric Acid Storage System level of 7% ensures there is a volume of greater than or equal to 740 (2652*) gallons available. An RWST level of 9% ensures there is a volume of greater than or equal to 38,740 gallons available.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The OPERABILITY of the automatic Boron Dilution Protection System ensures adequate capability for negative reactivity insertion to prevent a transient caused by the uncontrolled dilution of the RCS in MODES 3,4, and 5. The functioning of the system precludes the necessity of operator action to prevent further dilution by terminating flow to the charging pump(s) from possible unborated water sources and initiating flow from the RWST. The most restrictive condition occurs shortly after beginning of life when the critical boron concentration is highest, and a 205 gpm dilution flowrate provides the maximum positive reactivity addition rate. One reactor coolant pump in operation with all reactor coolant loop stop isolation valves open reduces the reactivity addition rate by mixing the dilution through all four reactor coolant loops. A minimum count rate of ten counts per second minimizes the impact of the uncertainties associated with the source range nuclear instrumentation. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3 Δk/k is required to control the reactivity transient. Actions taken by

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

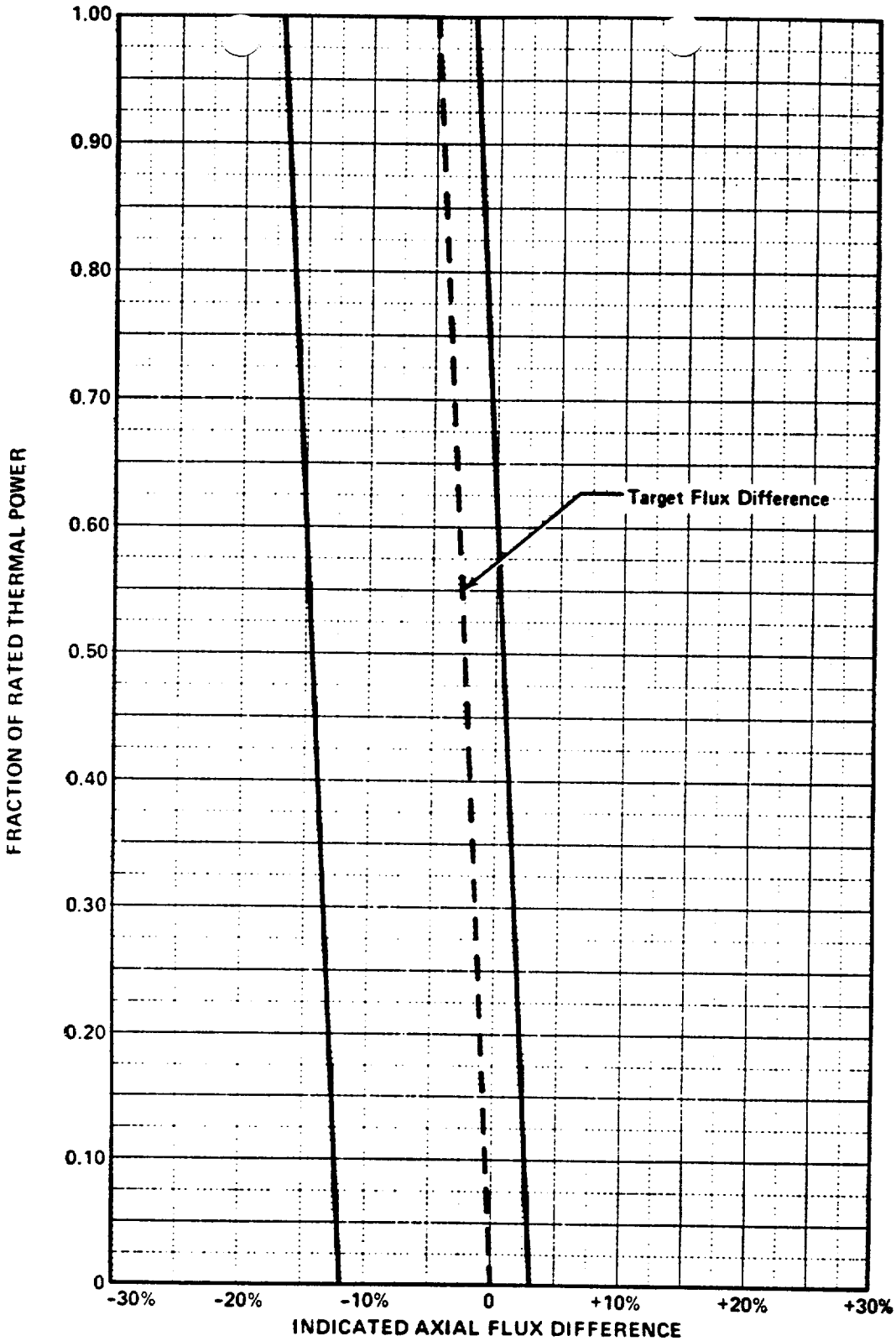

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

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

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

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

$F^N_{\Delta H}$ will be maintained within its limits provided the Conditions a. through d. above are maintained. The combination of the RCS flow requirement [371,400 gpm (390,400 gpm*)] and the requirement on $F^N_{\Delta H}$ guarantee that the DNBR used in the safety analysis will be met.

Margin between the safety analysis limit DNBRs [1.50 for the typical and thimble cells (1.49 and 1.47 for the OFA fuel typical and thimble cells, respectively and 1.67 and 1.65 for the VANTAGE 5 typical and thimble cells*)] and the design limit DNBRs [1.25 for the typical and thimble cells (1.34 and 1.32 for the OFA fuel typical and thimble cells, and 1.33 and 1.32 for the VANTAGE 5 fuel typical and thimble cells, respectively*)] is maintained.

A fraction of this margin is utilized to accommodate the transition core DNBR penalty (maximum of 12.5%) and the appropriate fuel rod bow DNBR penalty (less than 1.5% per WCAP-8691, Revision 1). The rest of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

The RCS flow requirement is based on the loop minimum measured flow rate of 92,850 gpm (97,600 gpm*) which is used in the Revised Thermal Design Procedure (Improved Thermal Design Procedure described in UFSAR 4.4.1 and 15.0.3*). A precision heat balance is performed once each cycle and is used to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% is assessed for potential feedwater venturi fouling. A maximum measurement uncertainty of 3.5% (2.2%*) has been included in the loop minimum measured flow rate to account for potential undetected feedwater venturi fouling and the use of the RCS flow indicators for flow rate verification. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken, before performing subsequent precision heat balance measurements, i.e., either the effect of fouling shall be quantified and compensated for in the RCS flow rate measurement, or the venturi shall be cleaned to eliminate the fouling.

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

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

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

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. A minimum contained borated water level of 89% ensures a volume of greater than or equal to 395,000 gallons.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

CONTAINMENT PURGE VENTILATION SYSTEM (Continued)

be exceeded in the event of an accident during containment purging operation. Operation with one line open will be limited to 1000 hours during a calendar year.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2.b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses. A spray additive tank level of between 78.6% and 90.3% ensures a volume of greater than or equal to 4000 gallons but less than or equal to 4540 gallons.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Cooling System ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 thru 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the Purge System) is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These Hydrogen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The Hydrogen Mixing Systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on Keff of no greater than 0.95 is sufficient to prevent reactor criticality during refueling operations and includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2300 ppm (2000 ppm*) or greater includes a conservative uncertainty allowance of 50 ppm. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

The Byron Station is designed such that the containment opens into the fuel building through the personnel hatch or equipment hatch. In the event of a fuel drop accident in the containment, any gaseous radioactivity escaping from the containment building will be filtered through the Fuel Handling Building Exhaust Ventilation System.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

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

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. 56
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 March 23, 1994, as supplemented on July 26, 1994, 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. 56 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: October 21, 1994

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. 55
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 March 23, 1994, as supplemented on July 26, 1994, 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. 55 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

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: October 21, 1994

ATTACHMENT TO LICENSE AMENDMENT NOS. 56 AND 55

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 by an asterisk are provided for convenience only.

Remove Pages	Insert Pages
III	III
2-1	2-1
2-2	2-2
-	2-2a
2-5	2-5
2-7	2-7
2-8	2-8
2-10	2-10
B 2-1	B 2-1
3/4 1-11	3/4 1-11
3/4 1-12	3/4 1-12
3/4 2-8	3/4 2-8
3/4 5-1	3/4 5-1
3/4 5-11	3/4 5-11
3/4 9-1	3/4 9-1
3/4 9-2*	3/4 9-2*
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 2-3*	B 3/4 2-3*
B 3/4 2-4	B 3/4 2-4
B 3/4 5-4	B 3/4 5-4
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4*	B 3/4 6-4*
B 3/4 9-1	B 3/4 9-1

SECTION PAGE

2.1 SAFETY LIMITS

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2.1.2 REACTOR COOLANT SYSTEM PRESSURE..... 2-1
FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION.. 2-2
FIGURE 2.1-1a REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION. 2-2a

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS..... 2-3
TABLE 2.2-1 REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS.... 2-4

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2.1 SAFETY LIMITS

2.1.1 REACTOR CORE..... B 2-1
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2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS..... B 2-3

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1* (Figure 2.1-1a)** for four loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within this limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within this limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

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

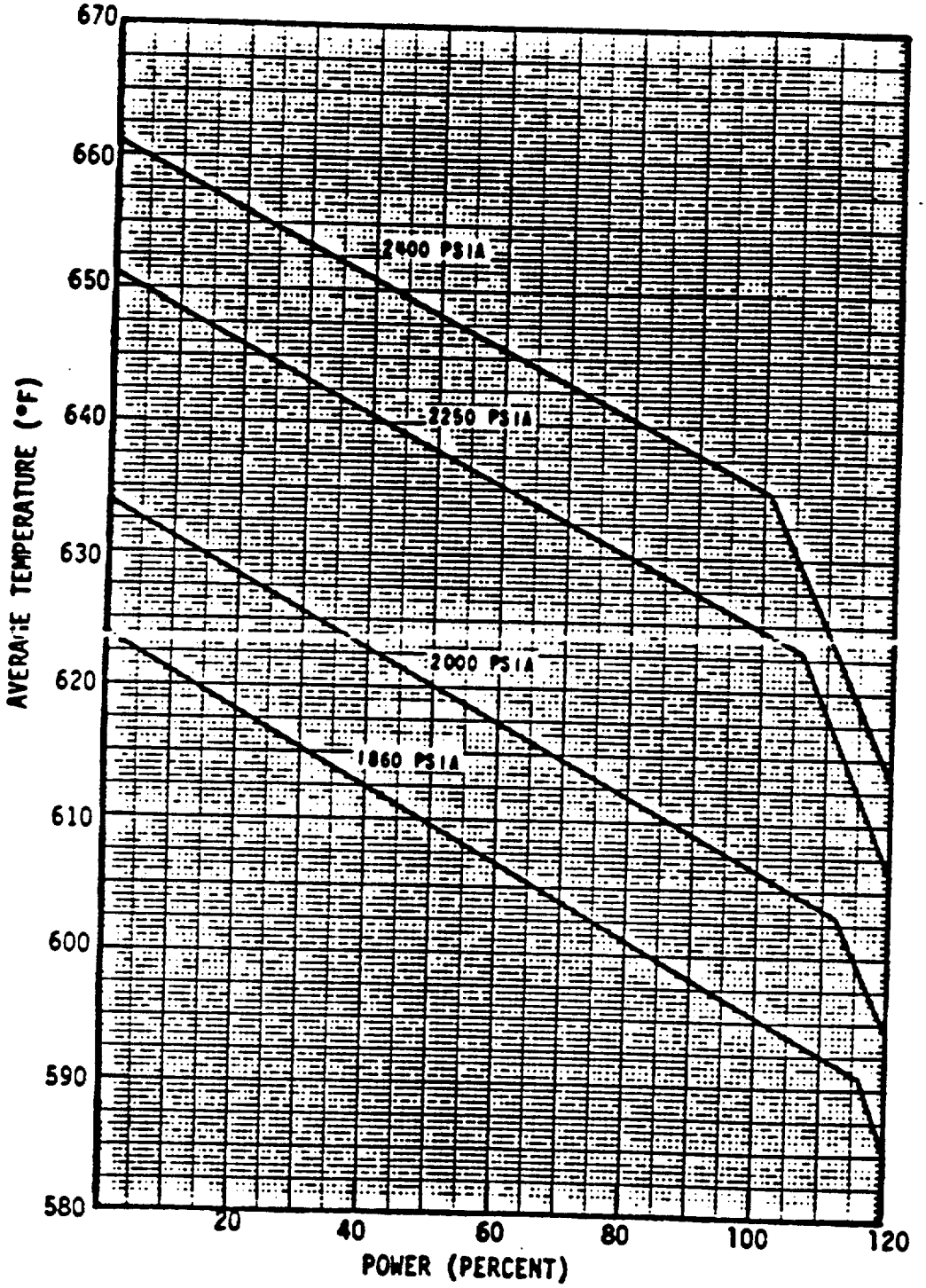

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

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

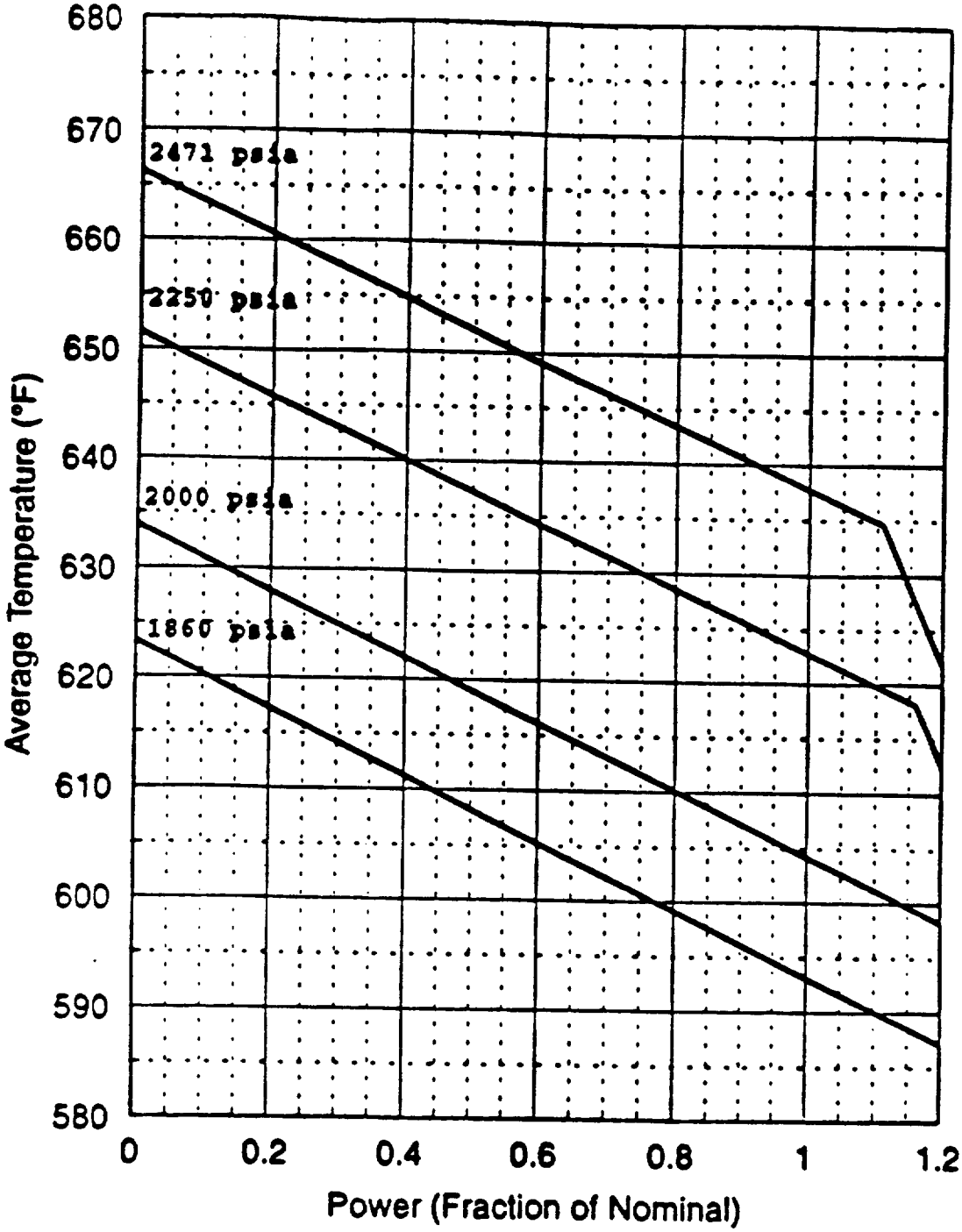

Figure 2.1-1a
Reactor Core Safety Limit - Four Loops
in Operation

Applicable to Unit 1 and Unit 2 starting with cycle 6

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** (92,850 gpm)#
**Applicable to Unit 1 and Unit 2 until completion of cycle 5.
#Applicable to Unit 1 and Unit 2 starting with cycle 6.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation,
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead-lag compensator on measured ΔT,
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s,
$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT,
τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3 = 0 s,
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER,
K_1	=	1.164,* 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.

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, $\tau_6 = 0$ s,

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

K_3 = 0.00134*, 0.00181**

P = Pressurizer pressure, psig,

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

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

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

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

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

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

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.71%*, 1.16%** 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%*, 3.08%** 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.

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNBR thermal design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95% (at a 95% confidence level) for any Condition I or II event.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered. As described in the UFSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

The design DNBR values are 1.34 and 1.32 for a typical cell and a thimble cell, respectively for OFA* fuel, and 1.33 for a typical cell and 1.32 for a thimble cell for the VANTAGE 5 fuel (1.25 for the typical and thimble cells)**. In addition, margin has been maintained in both designs by meeting safety analysis DNBR limits of 1.49 for a typical cell and 1.47 for a thimble cell for OFA fuel, and 1.67 and 1.65 for a typical cell and a thimble cell, respectively for the VANTAGE 5 fuel (1.50 for the typical and thimble cells)** in performing safety analyses.

The curves of Figure 2.1-1 (Figure 2.1-1a)** show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum design DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

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

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water level of 7.0%,

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water level of 9.0%,

2) a) *A minimum boron concentration of 2000 ppm,

b) **A boron concentration between 2300 and 2500 ppm, and

3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water level, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 35°F.

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

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2 for MODES 1, 2 and 3 and one of the following borated water sources shall be OPERABLE as required by Specification 3.1.2.1 for MODE 4:

a. A Boric Acid Storage System with:

1) A minimum contained borated water level of 40%,

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water level of 89%,

2) a) *A minimum boron concentration of 2000 ppm,

b) **A boron concentration between 2300 and 2500 ppm,

3) A minimum solution temperature of 35°F, and

4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources in MODE 1, 2, or 3, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable in MODE 1, 2, or 3, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With no borated water source OPERABLE in MODE 4, restore one borated water source to OPERABLE status within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

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

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 Indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows for four loop operation.

a. 1) *RCS Total Flowrate ≥ 390,400 gpm, and

2) **RCS Total Flowrate ≥ 371,400 gpm, and

b. $F^N_{\Delta H} \leq 1.55\ [1.0 + 0.3\ (1.0-P)]$ for OFA fuel

$F^N_{\Delta H} \leq 1.65\ [1.0 + 0.3\ (1.0-P)]$ for VANTAGE 5 fuel

where:

Measured values of $F^N_{\Delta H}$ are obtained by using the movable incore detectors. An appropriate uncertainty of 4% (nominal) or greater shall then be applied to the measured value of $F^N_{\Delta H}$ before it is compared to the requirements, and

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore RCS total flow rate and $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux-High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water level of between 31% and 63%,

c. 1) #A boron concentration between 1900 and 2100 ppm,

2) **A boron concentration between 2200 and 2400 ppm, and

d. A nitrogen cover-pressure of between 602 and 647 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water level and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.
#Applicable to Unit 1 and Unit 2 until completion of cycle 5.
**Applicable to Unit 1 and Unit 2 starting with cycle 6.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) and the heat traced portion of the RWST vent path shall be OPERABLE with:

a. A minimum contained borated water level of 89%,

b. 1) *A minimum boron concentration of 2000 ppm,

2) **A boron concentration between 2300 and 2500 ppm,

c. A minimum water temperature of 35°F, and

d. A maximum water temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water level in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 35°F or greater than 100°F, and

c. At least once per 24 hours by verifying the RWST vent path temperature to be greater than or equal to 35°F when the outside air temperature is less than 35°F.

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

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. A K_{eff} of 0.95 or less, or

b. 1) #A boron concentration of greater than or equal to 2000 ppm.

2) **A boron concentration of greater than or equal to 2300 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm# (2300 ppm)**, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 57 steps (approximately 3 feet) from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 Valves CV111B, CV8428, CV8441, CV8435, and CV8439 shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days.

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*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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

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

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. •A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} $\Delta k/k/°F$. The MTC value of -3.2×10^{-4} $\Delta k/k/°F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.1×10^{-4} $\Delta k/k/°F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC can be maintained within its limits. The BOL MTC measurement combined with the predicted MTC with core burnup can be used to impose administrative limits on rod withdrawal to ensure that MTC will always be less positive then 0 $\Delta k/k/°F$. This coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 550°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor vessel is above its minimum RT_{NDT} temperature, and (5) the plant is above the cooldown steam dump permissive, P-12.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement is 15,780 (13,487)* gallons of 7000-ppm borated water from the boric acid storage tanks or 70,450 (54,014)* gallons of 2000-ppm (2300-ppm)* borated water from the refueling water storage tank.

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

BORATION SYSTEMS (Continued)

A Boric Acid Storage System level of 40%* ensures that there is a volume of greater than or equal to 15,780 (13,487)* gallons available. A RWST level of 89% ensures that there is a volume of greater than or equal to 395,000 gallons available.

With the RCS temperature below 350°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 330°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or an RHR Suction relief valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2,652 (740)* gallons of 7000-ppm borated water from the boric acid storage tanks or 11,840 (2,264)* gallons of 2000-ppm (2300-ppm)* borated water from the refueling water storage tank (RWST). A Boric Acid Storage System level of 7% ensures there is a volume of greater than or equal to 2652 (740)* gallons available. An RWST level of 9% ensures there is a volume of greater than or equal to 38,740 gallons available.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The OPERABILITY of the automatic Boron Dilution Protection System ensures adequate capability for negative reactivity insertion to prevent a transient caused by the uncontrolled dilution of the RCS in MODES 3,4, and 5. The functioning of the system precludes the necessity of operator action to prevent further dilution by terminating flow to the charging pump(s) from possible unborated water sources and initiating flow from the RWST. The most restrictive condition occurs shortly after beginning of life when the critical boron concentration is highest, and a 205 gpm dilution flowrate provides the maximum positive reactivity addition rate. One reactor coolant pump in operation with all reactor coolant loop stop isolation valves open reduces the reactivity addition rate by mixing the dilution through all four reactor coolant loops. A minimum count rate of ten counts per second minimizes the impact of the uncertainties associated with the source range nuclear instrumentation. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3 Δk/k is required to control the reactivity transient. Actions taken by the microprocessor if the neutron count rate is doubled will prevent return to criticality in these MODES.

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

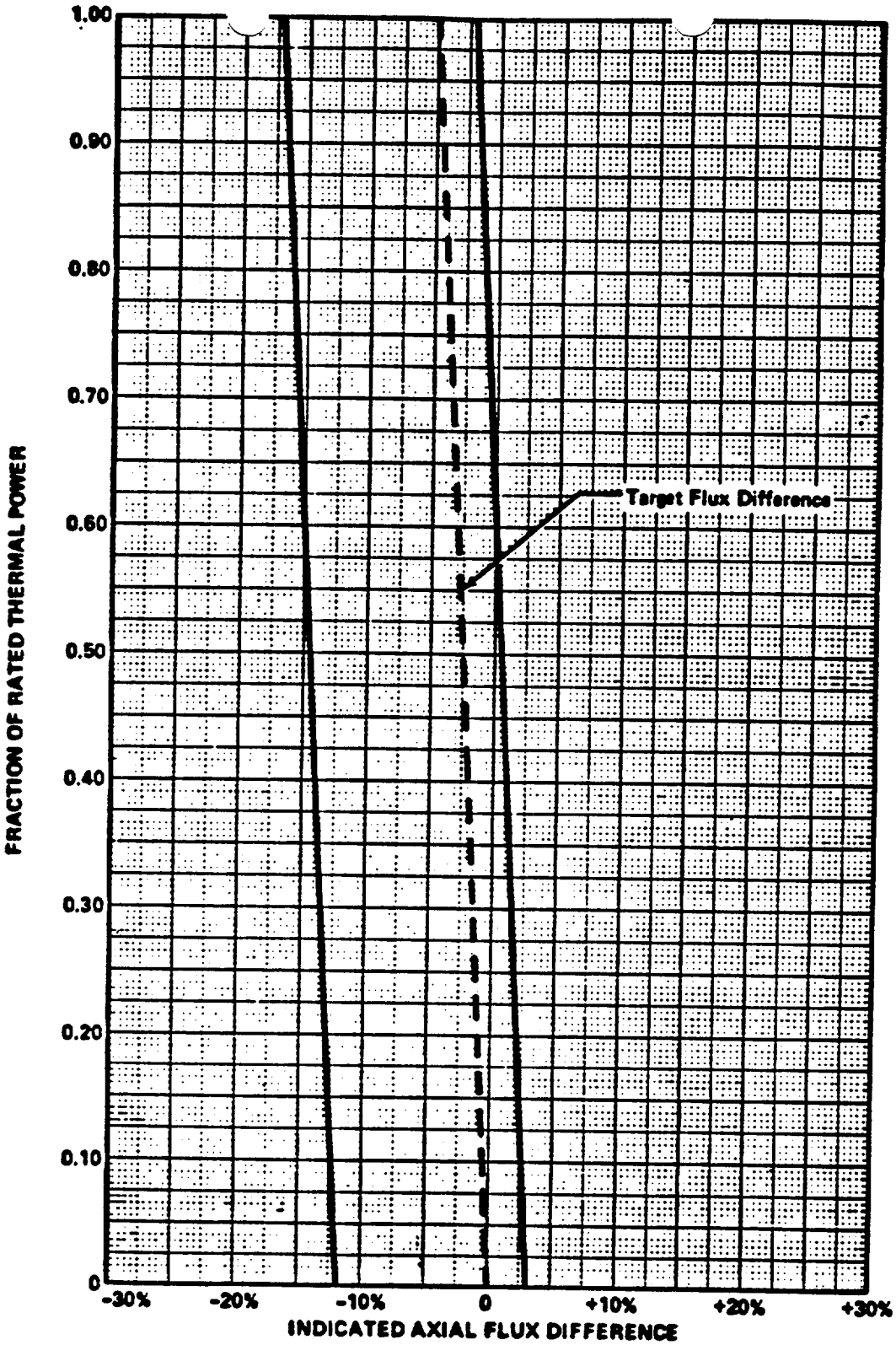

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

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

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

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

$F^N_{\Delta H}$ will be maintained within its limits provided the Conditions a. through d. above are maintained. The combination of the RCS flow requirement 390,400 gpm (371,400 gpm)* and the requirement on $F^N_{\Delta H}$ guarantee that the DNBR used in the safety analysis will be met.

Margin between the safety analysis limit DNBRs [1.49 and 1.47 for the OFA fuel typical and thimble cells, respectively and 1.67 and 1.65 for the VANTAGE 5 typical and thimble cells (1.50 for the typical and thimble cells)*] and the design limit DNBRs [1.34 and 1.32 for the OFA fuel typical and thimble cells, and 1.33 and 1.32 for the VANTAGE 5 fuel typical and thimble cells, respectively (1.25 for the typical and thimble cells)*] is maintained.

A fraction of this margin is utilized to accommodate the transition core DNBR penalty (maximum of 12.5%) and the appropriate fuel rod bow DNBR penalty (less than 1.5% per WCAP-8691, Revision 1). The rest of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

The RCS flow requirement is based on the loop minimum measured flow rate of 97,600 gpm (92,850 gpm)* which is used in the Improved Thermal Design Procedure (Revised Thermal Design Procedure)* described in UFSAR 4.4.1 and 15.0.3. A precision heat balance is performed once each cycle and is used to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% is assessed for potential feedwater venturi fouling. A maximum measurement uncertainty of 2.2% (3.5%)* has been included in the loop minimum measured flow rate to account for potential undetected feedwater venturi fouling and the use of the RCS flow indicators for flow rate verification. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken, before performing subsequent precision heat balance measurements, i.e., either the effect of fouling shall be quantified and compensated for in the RCS flow rate measurement, or the venturi shall be cleaned to eliminate the fouling.

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

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

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

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. A minimum contained borated water level of 89% ensures a volume of greater than or equal to 395,000 gallons.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

CONTAINMENT PURGE VENTILATION SYSTEM (Continued)

be exceeded in the event of an accident during containment purging operation. Operation with one line open will be limited to 1000 hours during a calendar year.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2.b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses. A spray additive tank level of between 78.6% and 90.3% ensures a volume of greater than or equal to 4000 gallons but less than or equal to 4540 gallons.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Cooling System ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 thru 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the Purge System) is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These Hydrogen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The Hydrogen Mixing Systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on Keff of no greater than 0.95 is sufficient to prevent reactor criticality during refueling operations and includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm (2300 ppm)* or greater includes a conservative uncertainty allowance of 50 ppm. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

The Braidwood Station is designed such that the containment opens into the fuel building through the personnel hatch or equipment hatch. In the event of a fuel drop accident in the containment, any gaseous radioactivity escaping from the containment building will be filtered through the Fuel Handling Building Exhaust Ventilation System.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 56 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 55 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

Commonwealth Edison Company (ComEd, the licensee), in a letter of March 23, 1994, as supplemented on July 26, 1994, proposed changes to the technical specifications (TS) for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. These changes involve TS requirements for a positive moderator temperature coefficient (PMTC) of up to +7 pcm per °C; and a reduction in the minimum core flow. As a consequence of these changes, TS changes are also proposed for the safety limit curves, safety system settings, boron concentration in various systems, and the Bases related to these TS, and to pH and departure from nucleate boiling (DNB). The proposed changes were to apply to the units as they enter their respective refueling outages beginning with Byron, Unit 1, which is scheduled to restart from its current refueling outage in November 1994.

The March 23, 1994, submittal included two reports: Commonwealth Edison Company Byron and Braidwood Units 1 and 2 Increased SGTP/Reduced TDF/PMTC Analysis Program Engineering/Licensing Report (WCAP 13964, Revision 1); and Operating Parameter Uncertainties for the Byron/Braidwood Revised Thermal Design Procedure, Revision 0.

The staff requested additional information on July 26, 1994, regarding the proposed changes to the MTC and received a response from the licensee dated August 16, 1994. The issues regarding approval of a PMTC are still under review by the staff. Consequently, this Safety Evaluation will address only the other requests included in the March 23, 1994, submittal.

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2.0 EVALUATION

The Westinghouse Electric Corporation's (W) SGTP/TDF/PMTC report provided W analyses and evaluations for Byron and Braidwood on: (1) the power capability parameters, (2) nuclear steam supply systems (NSSS), (3) NSSS design transients, (4) NSSS components, and (5) the relevant updated Final Safety Analysis Report (UFSAR) Chapter 15 events. Standard approval W procedures and methods applicable to the UFSAR were used. Complementary to that analysis is the uncertainty analysis that was also submitted. It followed the W Revised Thermal Design Procedure, that is standard W methodology and had been used in previous licensee analyses approved by the staff.

As indicated above, the proposed PMTC TS are still under review. However, analyses for future plant operations presented in the submittal include the possible use of the maximum proposed PMTC. This includes all transients and accidents and the determination of requirements for various boron concentrations in the accumulators and in the refueling water storage tank to accommodate reactivity changes associated with the requested PMTC. The boron requirements were increased in order to provide adequate shutdown margin throughout the cycle. The need for change would arise from potential increased cycle energy design with the PMTC. The requirements for shutdown margin and the necessary increased boron was included in the safety analyses performed by W. Thus, TS boron concentration values are generally increased from 2000 to up to 2300 ppm and the boration capability of systems increased. There are also changes to the TS Bases on pH values for relevant systems. Specific values and TS changes are discussed later. Calculations were done to verify post-LOCA conditions, involving pH and dose levels with maximum boron levels. The calculations and results for these shutdown margin and boron increase analyses follow normal procedures and are acceptable.

The current flow rate requirement included in the TS for Byron and Braidwood (390,400 gpm) is based on a minimum measured loop flow of 97,600 gpm. This is used in the thermal design procedure presented in the UFSARs. The licensee has proposed to lower the TS flow requirement to 371,400 gpm to provide a greater margin to actual flow, in anticipation of future steam generator tube plugging (SGTP). Associated with this change are proposed changes to TS for safety limits, safety system settings, flow rate requirements and the safety limit bases providing valves for departure from nucleate boiling ratio (DNBR) limits.

The W analysis program demonstrated that the steam generator tubes could be plugged to a maximum of 15 percent in any generator or to a reduction level of loop flow rate to the thermal design flow (TDF) limit of 89,700 gpm. The W analyses included related phenomena such as loop flow asymmetry of 5 percent, along with the proposed PMTC and boron increases. The analyses were based on previously approved T(hot) reduction programs for Byron and Braidwood. The core thermal limits were revised to include the use of the proposed flow changes and the use of the revised thermal design procedure. All systems affected by the change in core flow, plugging and associated parameters were reviewed and assessed by W for operation at the revised conditions. The

protection system setpoints were evaluated with the new reactor parameters and changed appropriately, and the transient and accident evaluations included these changes. Standard, approved W methods were used for these analyses, reviews and assessments. The results are acceptable.

Technical Specifications

The following TS and Bases changes were proposed for Byron and Braidwood. All changes followed from or were justified by the analyses presented in the submittal. Changes (2), (3) and (4), below, affecting the safety limit lines, the DNBR safety limit and trip system setpoint parameters, have occurred primarily because of the results of the uncertainty analysis of the Revised Thermal Design Procedure and the change in the core flow limit. The decreased flow limit (7) was proposed to provide more margin to the limit from actual flow rates to provide for future steam generator tube plugging. The proposed boron pH changes of (6), (8), (9), (10) and (11), which were included in the safety analyses, were made to accommodate increased cycle energy possible with cycle designs with the proposed PMTC. Without approval of the PMTC (5) at this time, the increased boron changes are conservative. With the exception of the PMTC change, the staff review determined that all proposed changes are acceptable. In the July 26, 1994, supplement to the original submittal, the licensee proposed additional footnotes to several of the changed TSs to address implementation of the changes on a unit and cycle specific basis. The original and supplemental footnotes, and indicated implementations are acceptable.

The changes to TS and Bases for both Byron and Braidwood from the original submittal are as follows:

1. Table of contents; Figure 3.1-0 added.

2. Figure 2.1-1, Reactor Core Safety Limits - Four Loops; limit lines changed.

3. Bases 2.1.1, Reactor Core; changed to provide new DNBR safety limits for OFA and VANTAGE 5 fuel.

4. Table 2.2-1, Reactor Trip System Instrumentation; flow limit and over temperature and over pressure delta T setpoints changed. Unit and cycle implementation noted.

5. TS 3.1.2.5 and 6, Borated Water Source-Shutdown and Operating; boron concentration changed to between 2,300 and 2,500 ppm, and implementation footnotes added.

6. TS 3.2.3, RCS Flow Rate; RCS total flowrate changed to 371,400 gpm.

7. TS 3.5.1, Accumulators; boron concentration changed to between 2200 and 2400 ppm and implementation footnote added.

8. TS 3.5.5, Refueling Water Storage Tank; boron concentration changed to between 2300 and 2500 ppm and implementation footnotes added.

9. TS 3.9.1, Refueling Operations/Boron Concentration; boron concentration changed to 2300 ppm.

10. Bases for the TS with flow or boron changes are changed to reflect the new values. Bases 3/4.6.22, Spray Additive System, has the lower pH limit changed from 8.5 to 8.0.

3.0 CONCLUSIONS

The staff has reviewed the information submitted by the licensee for Byron and Braidwood to justify proposed TS changes which provide for changes to the requirements or limits for the core flow as affected by steam generation tube plugging, and for associated changes to boron concentrations and pH level, safety system setpoints and DNBR safety limit values.

Based on this review, the staff concludes that appropriate information was submitted and the proposed changes to the TS are acceptable.

4.0 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.

5.0 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 effluent 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 (59 FR 41802). 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.

6.0 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: H. Richings

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