ATTACHMENT 2

PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS
FOR
FACILITY OPERATING LICENSES NPF-37, NPF-66, NPF-72, and NPF-77

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ATTACHMENT B
(continued)

BYRON

AFFECTED PAGES

Replace Figure 2.1-1 with Figure 2.1-1 on the next page

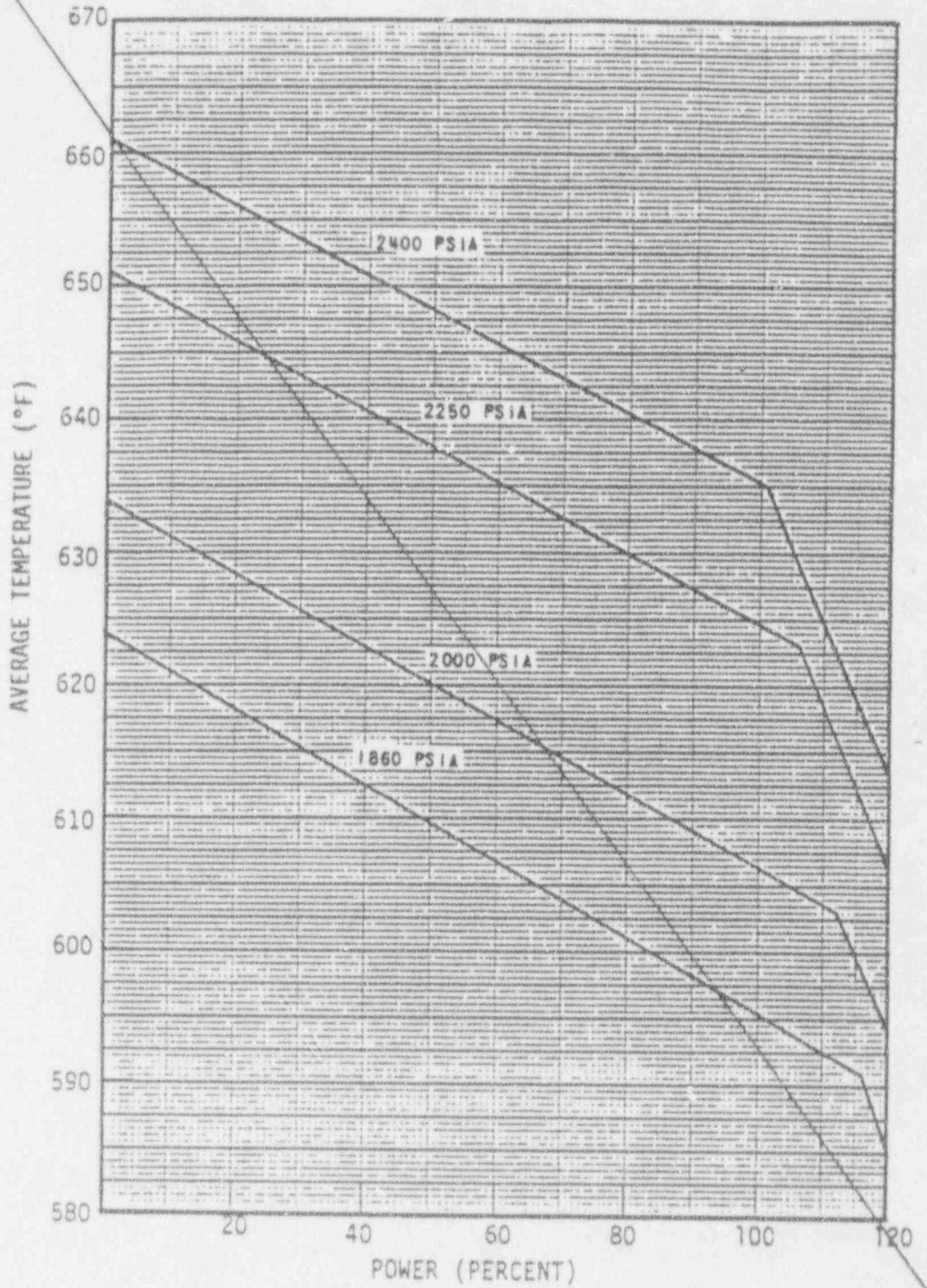

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

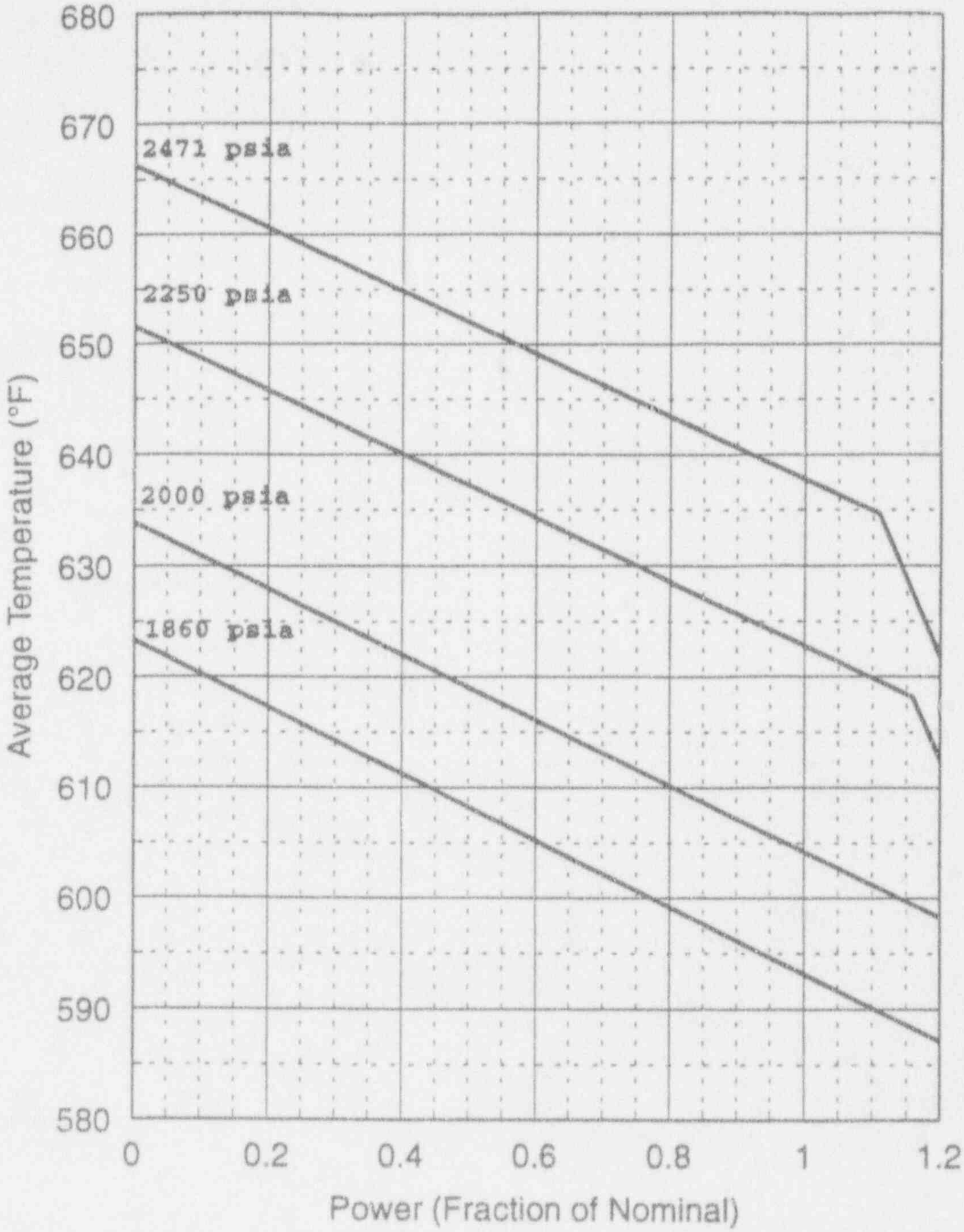

Figure 2.1-1
Reactor Core Safety Limit - Four Loops in Operation

2.1 SAFETY LIMITS

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.

SEE INSERT A

~~The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation for Optimized Fuel Assembly (OFA) fuel and the WRB-2 correlation for VANTAGE 5 fuel in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the correlation DNBR limit (1.17 for both the WRB-1 and WRB-2 correlations).~~

~~In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 confidence that the minimum DNBR for the limiting rods is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analysis using values of input parameters without uncertainties.~~

The design DNBR values are ~~1.34 and 1.32~~ 1.25 and 1.25 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 ~~1.49~~ 1.5 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 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.

Insert A

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.

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 = ~~97,600~~ gpm
92,850

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation,
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT,
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s,
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT,
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s,
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER,
1.325* K_1	=	(1.164)**
0.0297/°F* K_2	=	(0.0265/°F)**
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation,
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 = 33$ s, $\tau_5 = 4$ s,
T	=	Average temperature, °F,
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg},

* 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 1: (Continued)

τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s,
T^1	≤	588.4°F (Nominal T_{avg} at RATED THERMAL POWER),
K_3	=	0.00181* (0.00134)**
P	=	Pressurizer pressure, psig,
P^1	=	2235 psig (Nominal RCS operating pressure),
S	=	Laplace transform operator, s^{-1},

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

(i) for $q_t - q_b$ between -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)

0.00245/°F *

K_8 = (0.00170/°F)** for T > T" and K_8 = 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.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a.2** Less positive than 0 Δk/k/°F for the all rods withdrawn, hot zero THERMAL POWER condition, ~~or~~ and

b. Less negative than -4.1×10^{-4} Δk/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than (0 Δk/k/°F)## within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6; the limits of Figure 3.1-0**

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

a.1** Maintained within the limits specified in Figure 3.1-0,

*With K_{eff} greater than or equal to 1.

#See Special Test Exceptions Specification 3.10.3.

** 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.

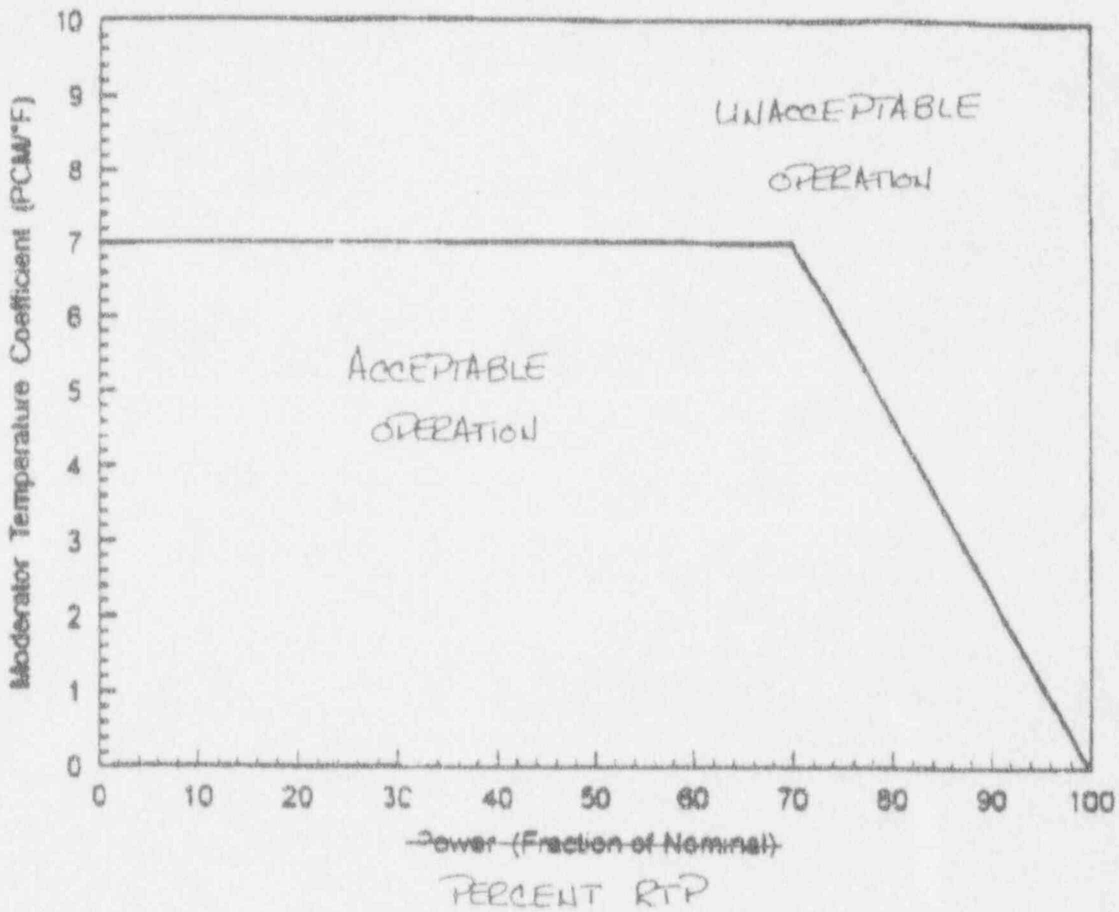

Figure 3.1-0 **

Moderator Temperature Coefficient vs. Power Level

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

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;
~~2)~~ 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.

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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.

BORATED WATER SOURCES - OPERATING

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) minimum contained borated water level of 89%,
2) a)* boron concentration between 2300 and 2500 ppm,
~~2)~~ b)** 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. RCS Total Flowrate $\geq$ ~~390,400 gpm~~ 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.

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,

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

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

~~c.~~ 2) **A boron concentration ~~of~~ 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.

EMERGENCY CORE COOLING SYSTEMS

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,

~~b.~~ 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.

* 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. A boron concentration of greater than or equal to ~~2000~~ 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~~ 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.

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} $\Delta k/k/^{\circ}F$. The MTC value of -3.2×10^{-4} $\Delta k/k/^{\circ}F$ 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} $\Delta k/k/^{\circ}F$.

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 $\Delta K/K/^{\circ}F$*. This coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

+7.0 x 10^{-5} $\Delta K/K/^{\circ}F$ for all rods withdrawn for power levels up to 70% RATED THERMAL POWER with a linear ramp to 0 $\Delta K/K/^{\circ}F$ at 100% RATED THERMAL POWER

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 ~~occurs at EOL from full power equilibrium xenon conditions and requires~~
13,487 (15,780)* gallons of 7000-ppm borated water from the boric acid storage tanks or
54,014 (70,450)* gallons of (2000-ppm)* borated water from the refueling water storage tank.
2300-ppm

* Not Applicable to Unit 1, Applicable to Unit 2 until completion of cycl

BORATION SYSTEMS (Continued) 13,487

A Boric Acid Storage System level of 40% ensures that there is a volume of greater than or equal to (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.

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
740 140°F. This condition requires either* (2,652) gallons of 7000-ppm borated
2264 water from the boric acid storage tanks or* (11,840) gallons of* (2000-ppm) borated 2300ppm
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) gallons 740 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 8.0 volume and boron concentration of the RWST also ensure a pH value of between ~~8.5~~ 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.

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

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 ~~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 ~~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 Revised on the loop minimum measured flow rate of 92,850 ~~97,600~~ gpm which is used in the ~~Improved~~ Thermal Design Procedure. ~~described in FSAR 4.4.1 and 15.0.3.~~ A precision heat balance is performed once each cycle and is used to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% is assessed for potential feedwater venturi fouling. A maximum measurement uncertainty of 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.

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 ~~8.5~~ 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 ~~8.5~~ 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.

3/4.9.1 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% Δk/k conservative allowance for uncertainties. Similarly, the boron concentration value of ~~2000~~ 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 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.

ATTACHMENT B
(continued)

BRAIDWOOD

AFFECTED PAGES

Replace Figure 2.1-1 with Figure 2.1-1 on next page

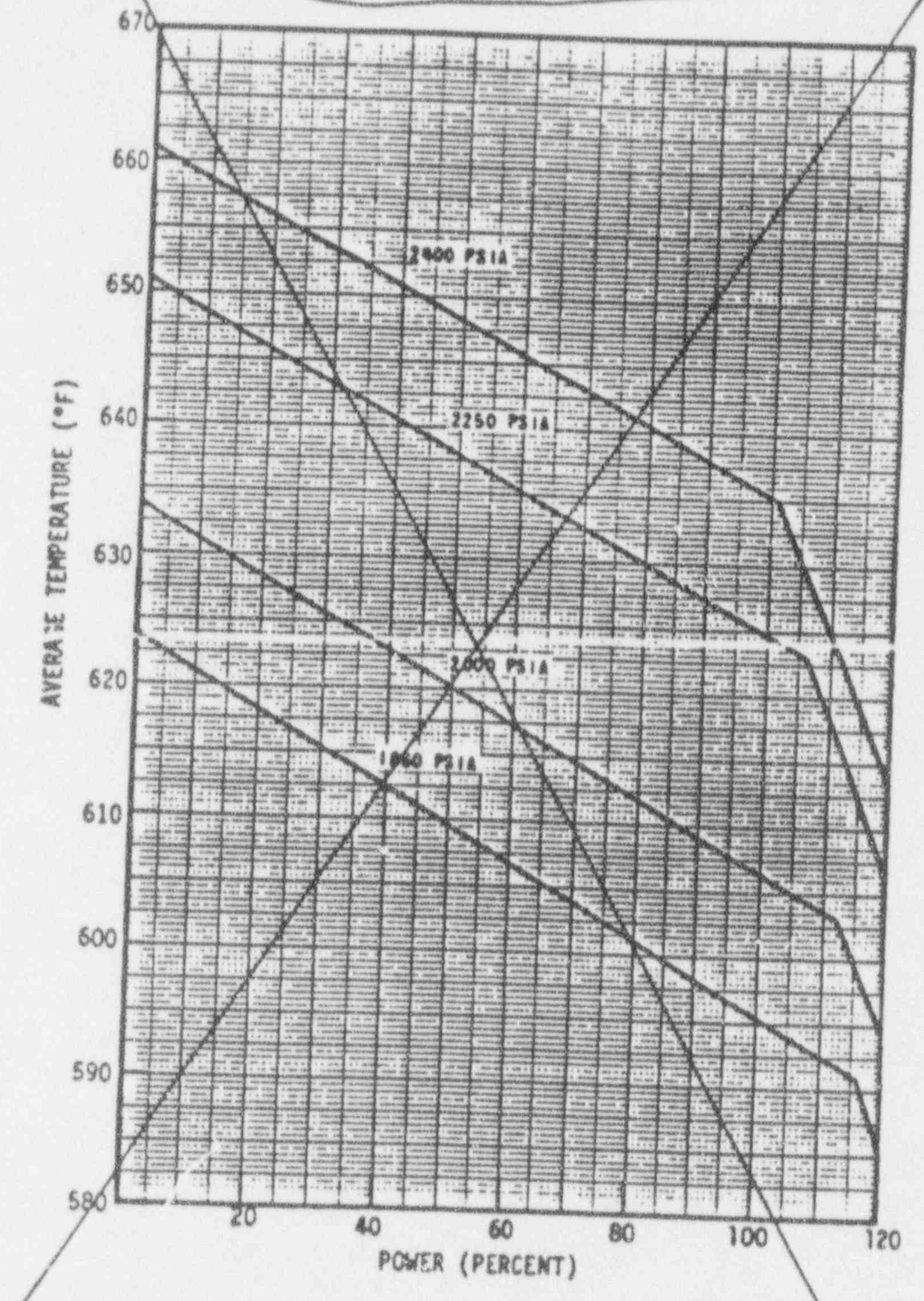

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

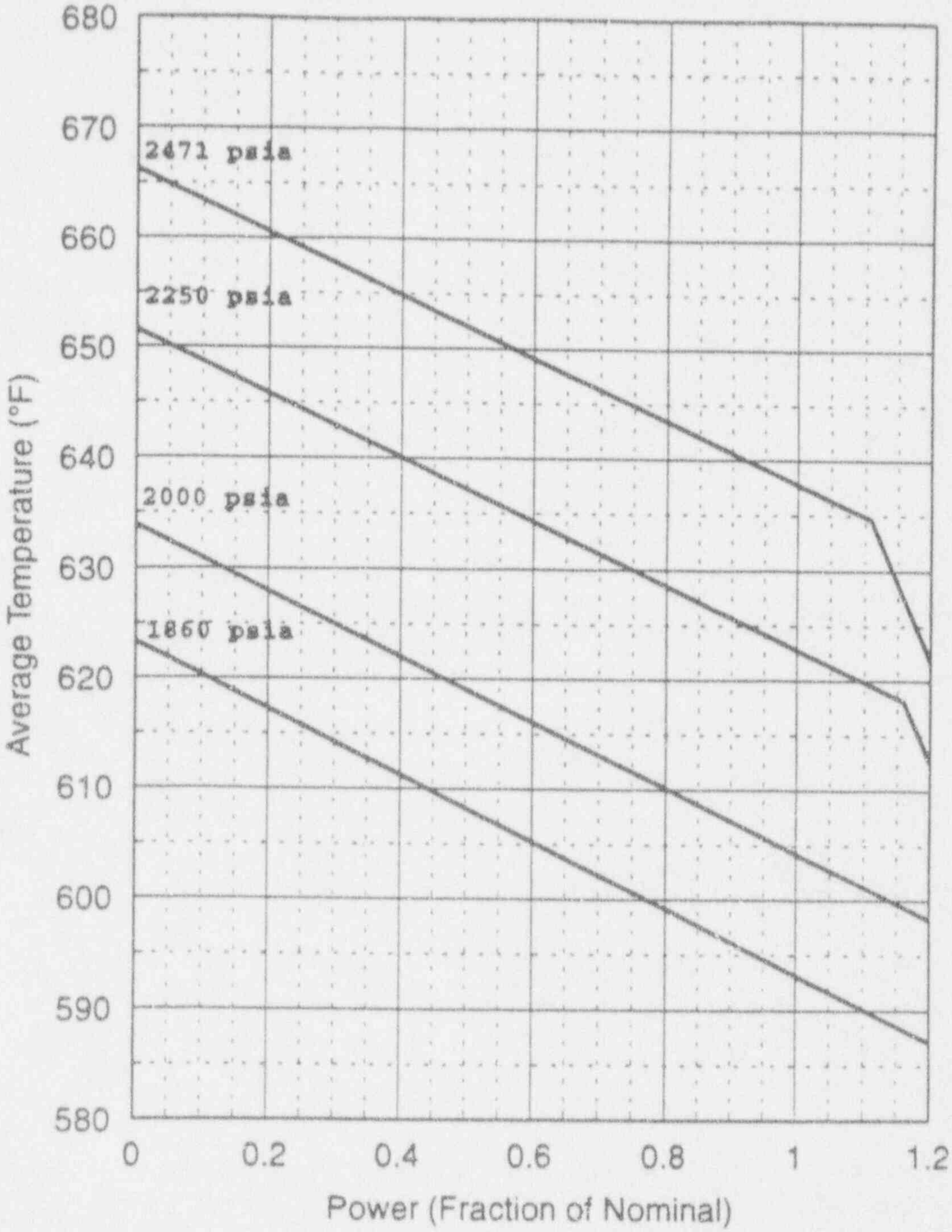

Figure 2.1-1
Reactor Core Safety Limit - Four Loops in Operation

2.1 SAFETY LIMITS
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.

INSERT A → ~~The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation for OFA* fuel and the WRB-2 correlation for VANTAGE 5 fuel in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the correlation DNBR limit (1.17 for both the WRB-1 and WRB-2 correlations).~~

~~In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 confidence that the minimum DNBR for the limiting rods is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analysis using values of input parameters without uncertainties.~~ The design DNBR values are ~~1.34~~ 1.25 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 ~~1.49~~ 1.50 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 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~~

Insert A

The DNBR thermal design criterion is the probability that DNB will not occur on the most limiting rod and 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.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	$\geq$90% of loop minimum measured flow*	$\geq$89.3% of loop minimum measured flow*
13. Steam Generator Water Level Low-Low		
a. Unit 1	$\geq$33.0% of narrow range instrument span	$\geq$31.0% of narrow range instrument span
b. Unit 2	$\geq$17% (Cycle 3); $\geq$36.3% (Cycle 4 and after) of narrow range instrument span	$\geq$16.3% (Cycle 3); $\geq$34.8% (Cycle 4 and after) of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	$\geq$5268 volts - each bus	$\geq$4920 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	$\geq$57.0 Hz	$\geq$56 08 Hz
16. Turbine Trip		
a. Emergency Trip Header Pressure	$\geq$1000 psig	$\geq$815 psig
b. Turbine Throttle Valve Closure	$\geq$1% open	$\geq$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

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

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

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

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

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

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

T' $\leq$ 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%* 3.33%** 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.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a.1.** Less positive than 0 Δk/k/°F for the all rods withdrawn, hot zero THERMAL POWER condition, ~~or~~ and

Insert B →

b. Less negative than -4.1×10^{-4} Δk/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 Δk/k/°F** (the limit of Figure 3.1-0)## within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.

#See Special Test Exceptions Specification 3.10.3.

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

##- Applicable to Unit 1 and Unit 2 starting with cycle 6.

Insert B

a.2.## Maintained within the limits specified in FIGURE 3.1-0, and

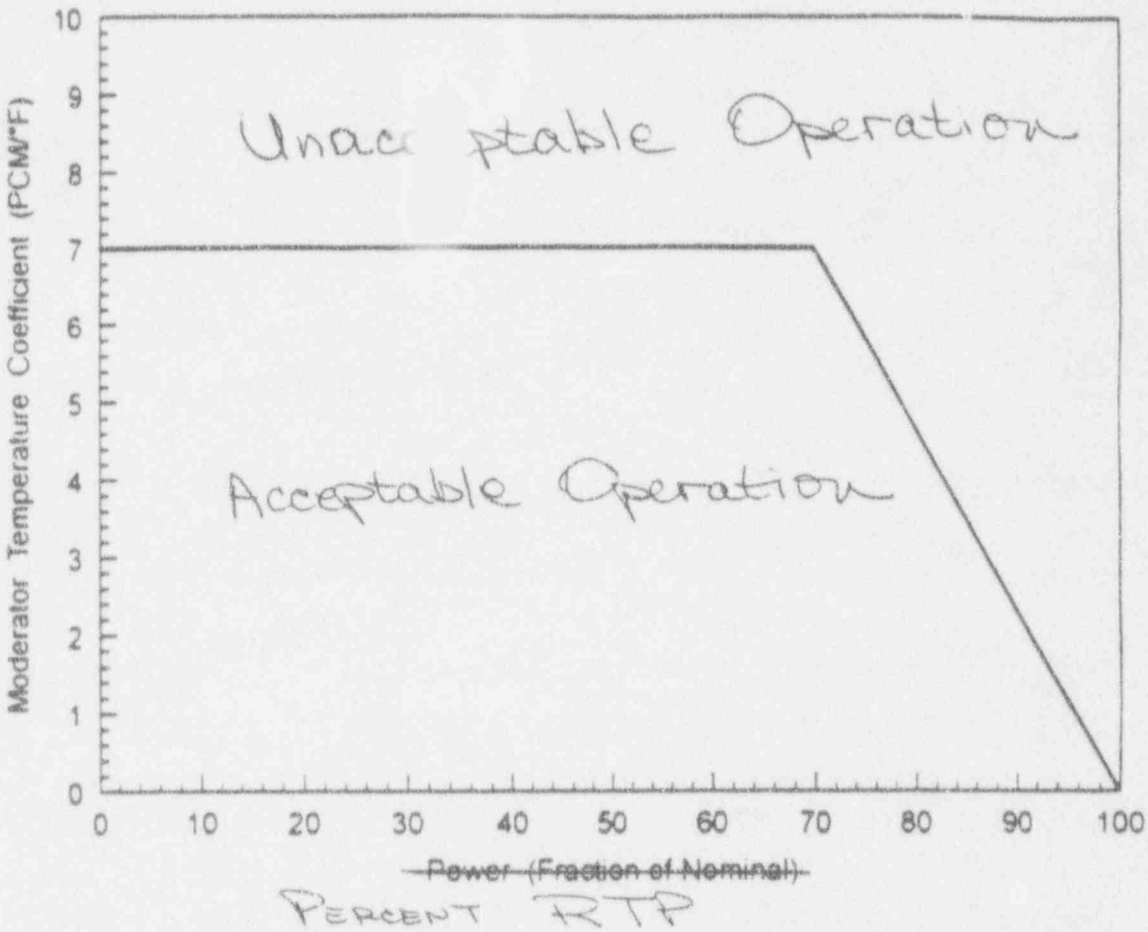

FIGURE 3.1-0**

Moderator Temperature Coefficient vs Power Level

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

REACTIVITY CONTROL SYSTEMS

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, and
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.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

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% Δk/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. RCS Total Flowrate ≥ 371,400 ~~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.

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 ~~of~~ between 1900 and 2100 ppm, and
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.

EMERGENCY CORE COOLING SYSTEMS

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.

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} Δk/k/°F. The MTC value of -3.2×10^{-4} Δk/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} Δk/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 ΔK/K/°F (+7.0 x 10^{-5} Δk/k/°F for all rods withdrawn for power levels up to 70% RATED THERMAL POWER with a linear ramp to 0 Δk/k/°F at 100% RATED THERMAL POWER)* 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% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement is ~~occurs at EOL from full power equilibrium xenon conditions and requires~~ (13,487)* 15,780 gallons of 7000-ppm borated water from the boric acid storage tanks or (54,014)* 70,450 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% $\Delta k/k$ 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 ~~8.5~~ 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 $\Delta k/k$ 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.

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.

$F_{\Delta H}^{N}$ will be maintained within its limits provided the Conditions a. through d. above are maintained. The combination of the RCS flow requirement (~~390,400~~ 376400 gpm) and the requirement on $F_{\Delta H}^{N}$ guarantee that the DNBR used in the safety analysis will be met.

Margin between the safety analysis limit DNBRs (1.50 ~~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 ~~(1.34 and 1.32 for the OFA fuel typical and thimble cells, and 1.33 and 1.32~~ 1.25 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 ~~97,600~~ 92,850 gpm which is used in the ~~Improved~~ Revised Thermal Design Procedure. ~~described in FSAR 4.4.1 and 15.0.3.~~ A precision heat balance is performed once each cycle and is used to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% is assessed for potential feedwater venturi fouling. A maximum measurement uncertainty of ~~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.

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 ~~8.5~~ 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 ~~0.5~~ 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.

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% Δk/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.

ATTACHMENT 3
EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

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

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

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

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

The proposed changes would modify the Technical Specifications concerning (1) the moderator temperature coefficient (MTC), (2) the boron concentration necessary to meet shutdown margin (SDM) requirements, and (3) the thermal design flowrate.

The MTC change would allow a slightly positive MTC (PMTC) below 100 percent of rated full power. The principal benefit of this change is that it would facilitate the design of future reload fuel cycles. Technical Specification changes are also required to meet SDM requirements to accommodate the positive MTC and the potential of lengthened reload fuel cycles due to increased energy requirements. To assure subcriticality requirements are met following a postulated loss-of-coolant accident (LOCA), the boron concentration is increased for the refueling water storage tank (RWST) and the accumulators. The safety analyses for the Byron and Braidwood Updated Final Safety Analysis Report (UFSAR) transients have been previously based on a maximum MTC being less than or equal to 0 pcm/°F at all times when the reactor is critical. The proposed change to the Technical Specification would allow a +7 pcm/°F MTC for power levels up to 70 percent with a linear ramp to 0 pcm/°F at 100 percent power. CECo has reviewed the revised USFAR safety analyses which conservatively bounds this positive MTC, increase in boron concentration, incorporates revised thermal design flows, and addresses increased tube plugging levels. The results of the revised analyses are provided in WCAP 13964 "Commonwealth Edison Company Byron and Braidwood Units 1 and 2 Increased SGTP/Reduced TDF/PMTC Analysis Program Engineering/Licensing Report".

The thermal design flow (TDF) is a minimum RCS flow value assumed in the accident analyses and reactor core thermal/hydraulic design calculations that demonstrate the necessary heat removal from the core to meet various transient acceptance criteria. The minimum measured flow (MMF) currently used for the licensing basis is a total core flow of 390,400 gpm for Byron/Braidwood and is reflected in Technical Specification Table 2.2-1 (Functional Unit 12) as a footnote of 97,600 gpm per loop for the reactor coolant flow-low reactor trip. The MMF value must be verified in accordance with Technical Specification 3/4.2.3.

A reduction in TDF has been factored into the accident analyses that rely on RCS flowrate. This results in a reduction in the limiting condition for operation (LCO) value for RCS flow reflected in the Technical Specifications. The reduced flow requirement provides a margin to account for steam generator tube plugging (SGTP). The revised TDF value corresponds to a MMF value of 371,400 gpm, which assumes a 3.5 percent flow measurement allowance, and is reflected in the footnote to Technical Specification Table 2.2-1 as 92,850 gpm, minimum measured loop flow for the reactor coolant flow-low reactor trip. The revised LCO flow value shall be incorporated in Technical Specification 3/4.2.3.

The proposed changes also include an administrative change to correct the wording in the MTC Technical Specification LCO 3.1.1.3a to clarify that both LCO 3.1.1.3a and b must be met over the fuel cycle.

Based on Commonwealth Edison's review and approval of WCAP 13964, which used NRC approved safety analysis methodology provided by Westinghouse and Commonwealth Edison, it has been determined that the changes associated with the analyses do not involve a significant hazard. Specifically:

A. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(1) The reduced thermal design flow and positive moderator temperature coefficient program, which includes corresponding changes to the RWST and accumulator required boron concentration, will not affect the operability and integrity of plant systems and components. The analysis program does not result in a condition where the design, material, and construction standards that were applicable prior to application of the program are altered. Additionally, the safety functions of the evaluated systems and components have not changed. The safety analyses necessary to support the reduced TDF and PMTC program were performed (WCAP 13964) and found to be acceptable and consistent with the Byron and Braidwood original safety analysis bases. All Departure from Nucleate Boiling (DNB) Ratio (DNBR) design limits were determined such that there was a 95 percent probability at a 95 percent confidence level that a DNBR value of 1.25 for a typical and thimble cell were verified to have been met. The present Technical Specification limit for Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$, of less than 1.65 ensures that the limiting DNB ratio during normal operations and operational transients (Condition I and Condition II events) is greater than

or equal to the DNBR limit of the correlation being applied.

The accidents which are found to be sensitive to PMTC were analyzed as part of this effort and the results were found to be acceptable. On a cycle-by-cycle basis, the impact of PMTC on Anticipated Trip Without Scram (ATWS) risk will be addressed by determining the Unfavorable Exposure Time (UET) per established Westinghouse Owners Group methodology, with corrective actions to be taken as appropriate to assure acceptable risk. The increase in the RWST and accumulator boron concentration will have no adverse impact on the previously evaluated accidents. The SGTP/TDF/PMTC program does not affect the integrity of the safety related systems and components such that their function to control radiological consequences is affected and all fission barriers will remain intact. The effects on offsite doses have been considered. The incorporation of a PMTC, a reduction in TDF and increased tube plugging levels have increased offsite doses. However, the increases are small and the total doses are a small fraction of the 10 CFR 100 limits. As such, the acceptance criteria continue to be satisfied. Therefore, the probability or consequences of an accident previously analyzed in the UFSAR is not increased by the SGTP/TDF/PMTC program.

B. The proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

(2) The methodology and manner of plant operation as a result of the proposed changes is unchanged. The increased SGTP, reduced TDF, and PMTC program, which includes changes to the RWST and accumulator boron concentration, does not impact the safe operation of the reactor provided that the existing and proposed Limiting Conditions for Operation (LCOs) and the associated action requirements are satisfied. The assumptions do not create failure modes that could adversely impact safety related equipment. The related Safety Limits and LCOs in the plant Technical Specifications will be addressed and evaluated for each reload core design via the 10 CFR 50.59 process. All DNBR design limits were determined such that there was a 95 percent probability at a 95 percent confidence level that a design DNBR value of 1.25 for a typical and thimble cell were verified to have been met. Other than the analysis for tube plugging, the proposed changes do not involve any equipment additions or modifications at the stations. Currently installed equipment will not be operated in a manner different than previously operated. Changes will be made to technical data within the existing station procedures, however, the analytical methods used to determine the data will remain unchanged. All aspects of the

SGTP/TDF/PMTC program have been evaluated, and no new or different accidents or failure modes have been identified for any system or component important to safety. Also, no new credible limiting single failure has been created. Because the SGTP/TDF/PMTC program does not adversely affect the integrity of the steam generator or any other equipment, it is determined that an accident different than any evaluated in the UFSAR will not be created.

C. The proposed changes do not involve a significant reduction in a margin of safety.

(3) The performance and integrity of the evaluated safety-related systems and components are not affected such that their control of radiological consequences is altered. The reduced TDF and PMTC program, which includes changes to the RWST and Accumulator boron concentration, will have no effect on the availability, operability, or performance of the evaluated safety-related systems or components. The margin of safety associated with the licensing basis safety analysis is not reduced by the changes. All acceptance criteria for the specific UFSAR Chapter 15 safety analyses (Non-LOCA and LOCA) have been either evaluated or verified to be met using NRC approved methodologies. Therefore, there is no significant reduction in the margin of safety as defined in the bases to any Technical Specification.

Based on the above evaluation, Commonwealth Edison has concluded that implementation of a PMTC, revised RWST and accumulator boron concentrations, and reduced RCS thermal design flow does not involve significant hazards consideration with respect to the provisions of 10CFR50.92.

ATTACHMENT 4
ENVIRONMENTAL ASSESSMENT

Commonwealth Edison has evaluated the proposed changes associated with the SGTP/TDF/PMTC program against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21. It has been determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth 10CFR51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10CFR50, it involves changes to a surveillance requirement and the amendment meets the following specific criteria:

(i) the amendment involves no significant hazards consideration.

As demonstrated in Attachment 3, this proposed amendment does not involve any significant hazards considerations.

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

The effects on offsite doses have been considered. The incorporation of a PMTC, a reduction in TDF and increased tube plugging levels have increased offsite doses. However, the increases are small and the total doses are a small fraction of the 10 CFR 100 limits. As such, the acceptance criteria continue to be satisfied. The proposed program assumptions do not change, degrade, or prevent the response of the evaluated safety-related systems and components such that their function in the control of radiological consequences is affected.

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

This proposed change will not result in changes in the operation or configuration of the facility; there will be no change in the level of controls or methodology used for processing of radioactive effluents of handling of solid radioactive waste nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

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

ATTACHMENT 5

WCAP 13964
Revision 1

Commonwealth Edison Company Byron and Braidwood Units 1 and 2 Increased SGTP/Reduced TDF/PMTC Analysis Program Engineering/Licensing Report