ATTACHMENT 2

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

BYRON STATION UNITS 1 & 2	BRAIDWOOD STATION UNITS 1 & 2
REVISED PAGES:	REVISED PAGES:
III	III
2-1	2-1
2-2	2-2
2-2a	2-2a
B 2-1	B 2-1
2-5	2-5
3/4 2-8	3/4 2-8
B 3/4 2-4	B 3/4 2-4
3/4 9-1	
B 3/4 9-1	

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AFFECTED PAGES

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECTION PAGE

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE.. 2-1

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

BASES

SECTION PAGE

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE.. B 2-1

2.1.2 REACTOR COOLANT SYSTEM PRESSURE............................. B 2-2

2.2 LIMITING SAFETY SYSTEM SETTINGS

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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* for four loop operation. (Figure 2.1-1a**)

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. Applicable to Unit 2 after cycle 5.

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

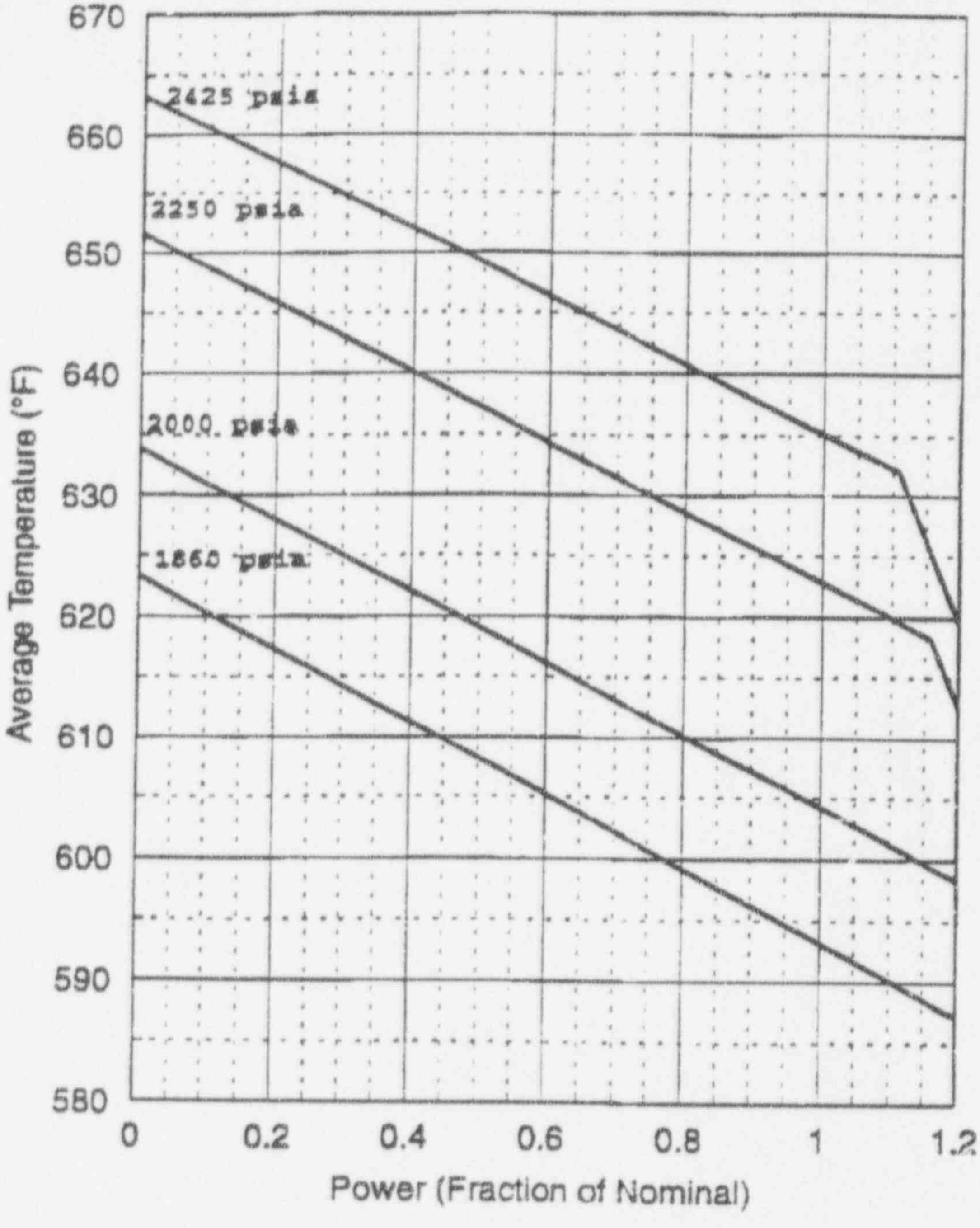

Figure 2.1-1
Reactor Core Safety Limit - Four Loops in Operation

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

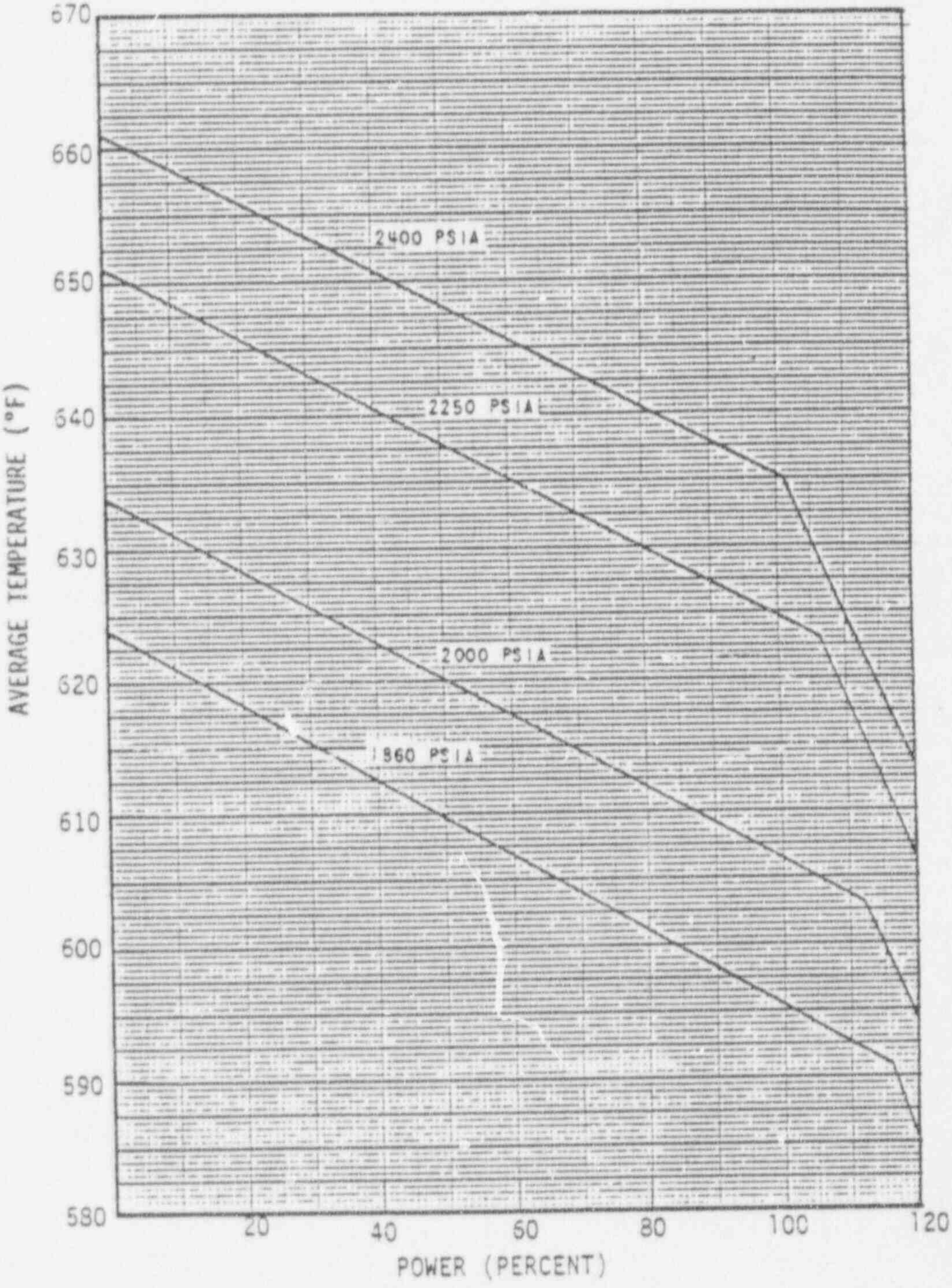

FIGURE 2.1-1a

REACTOR CORE SAFETY LIMIT - FOUR LOOPS
IN OPERATION

Not Applicable to Unit 1. Applicable to Unit 2 until completion

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 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 1.25 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 cell (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

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 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.

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,

~~a.~~ 2)** RCS Total Flowrate $\geq$ 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.

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.

[371,400gpm,

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

[1.50 for the typical and thimble cells,

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) *] 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) *] is maintained.

[1.25 for the typical and thimble cells

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.

92,856gpm

Revised Thermal Design Procedure

The RCS * flow requirement is based on the loop minimum measured flow rate of (97,600 gpm) which is used in the Improved Thermal Design Procedure described in ~~FSAR~~ 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.

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

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.

*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.

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 2300ppm (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.

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

BRAIDWOOD

AFFECTED PAGES

SECTION PAGE

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE.. 2-1

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

BASES

SECTION PAGE

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE.. B 2-1

2.1.2 REACTOR COOLANT SYSTEM PRESSURE............................ B 2-2

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.

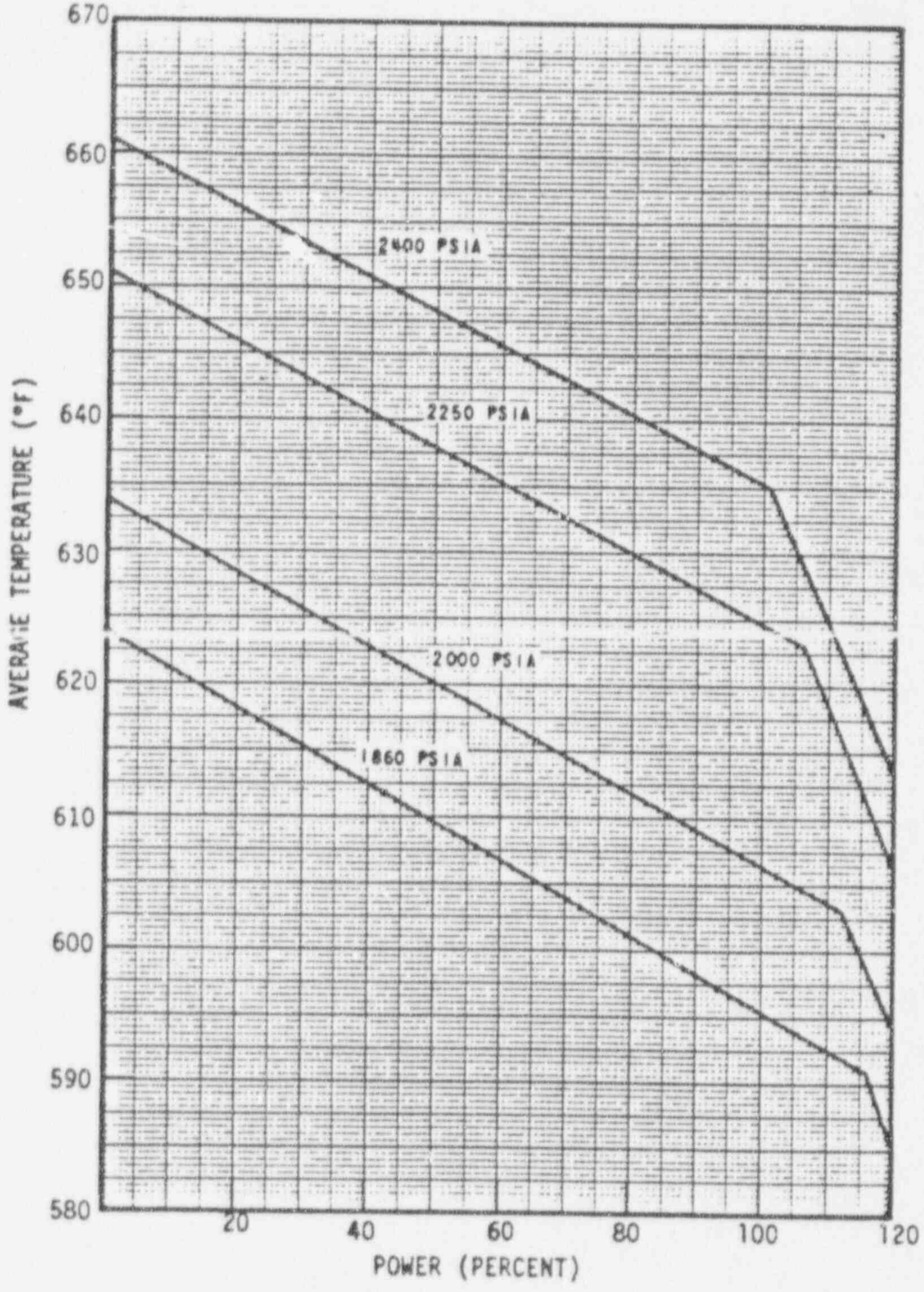

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

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

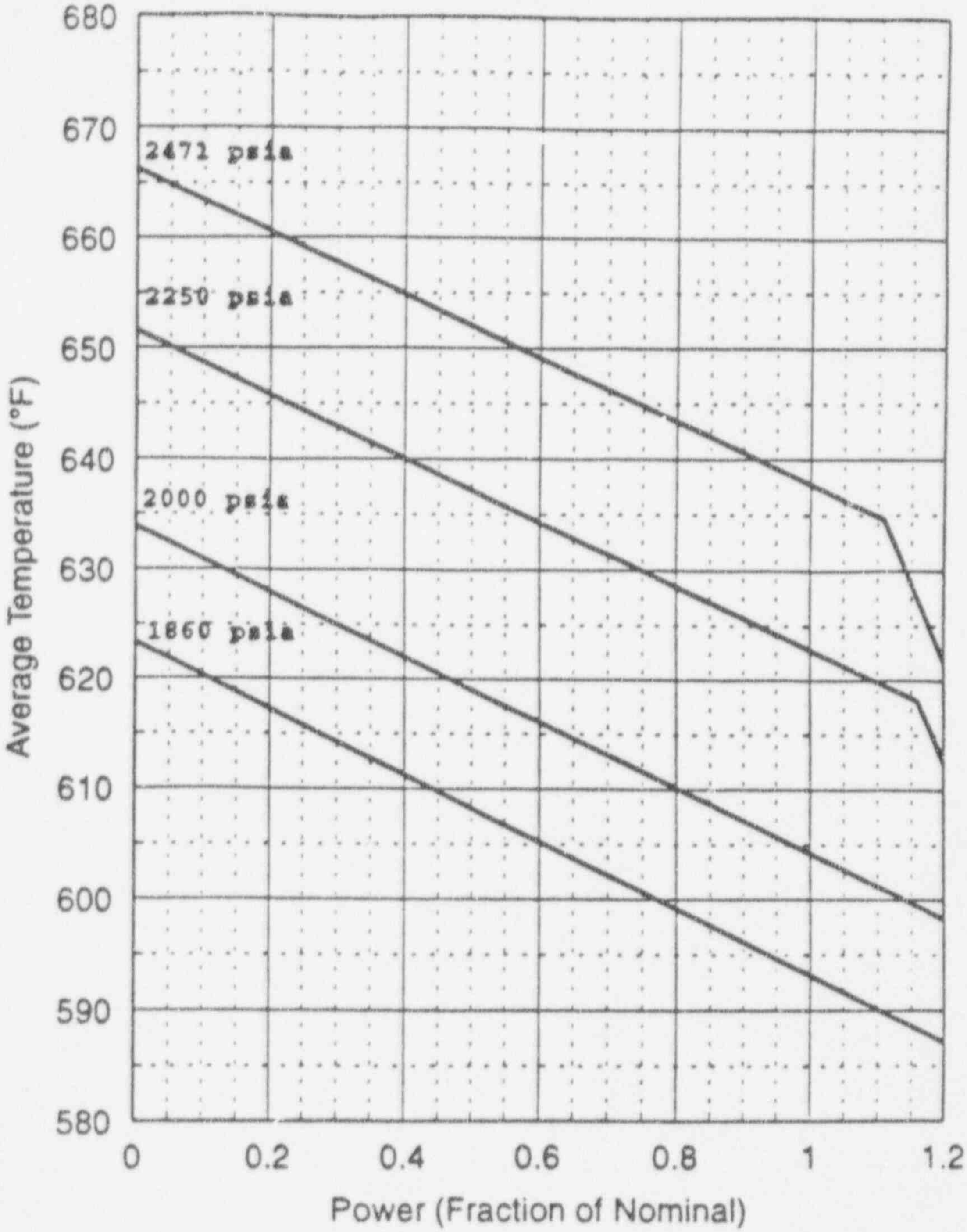

Figure 2.1-1a
Reactor Core Safety Limit - Four Loops in Operation

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

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).~~ (1.25 for the typical and thimble cells)**

~~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 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 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.

(1.50 for the typical and thimble cells)**

*Optimized Fuel Assemblies

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

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 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** (92,850gpm)#

** 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.

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.

(371,400 gpm)*

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

(1.50 for the typical and thimble cells)*

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 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 is maintained.

(1.25 for the typical and thimble cells)*

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.

(92,850 gpm)* (Revised Thermal Design Procedure)*

The RCS flow requirement is based on the loop minimum measured flow rate of 97,600 gpm which is used in the Improved Thermal Design Procedure described in ~~FSAR~~ 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.

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 to yield significant fuel cost savings. 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 in 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. These analyses conservatively bound the positive MTC and increase in boron concentration, incorporates the revised thermal design flows, and addresses increased tube plugging levels. The acceptable 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. These calculations demonstrate that the necessary heat is removed 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. This value 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 continues to provide a margin to account for future 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. This value 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 will 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.

WCAP 13964 utilized a NRC approved safety analysis methodology. Based on Commonwealth Edison's review and approval of WCAP 13964, 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 includes corresponding increases to the RWST and accumulator required boron concentration. The analysis program and associated boron concentration changes will not affect the operability and integrity of plant systems and components. The analysis program also does not result in a condition that would challenge the design, material, and construction standards of the plant systems and components. Additionally, the safety functions of the evaluated systems and components remain unchanged. 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 thus fuel integrity will not be challenged.

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 will result in a small increase in offsite doses, however, the total doses remain a small fraction of the 10 CFR 100 limits. As such, the accident analysis 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 reactor response to normal temperature fluctuations will be different due to PMTC, however, the normal reactor control systems, as designed, will continue to maintain a stable primary system temperature and reliable power production. The assumptions do not create any new failure modes that could adversely impact safety related equipment. The related Safety Limits and LCOs in the plant Technical Specifications will be evaluated and satisfied for each future reload core design via the 10 CFR 50.59 process. All DNBR Limits have been satisfied. The typical and thimble fuel cells were verified to maintain a DNBR value of 1.25 at a 95 percent probability and 95 percent conference level. 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 designed. Changes will be made to technical data within the existing

station procedures, however, the analytical methods used to determine the data also 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. 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 the proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

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 by the proposed changes. The radiological consequences of all previously analyzed accidents remain unchanged. 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 reactor response to normal temperature fluctuations will be different due to PMTC, however, the normal reactor control systems, as designed, will continue to maintain a stable primary system temperature and reliable power production. The margin of safety associated with the licensing basis safety analysis is not significantly reduced by the proposed changes. All acceptance criteria for the specific UFSAR Chapter 15 safety analyses (Non-LOCA and LOCA) have been satisfactorily evaluated and verified using NRC approved methodologies. Therefore, there is no significant reduction in the margin of safety as defined in the bases of any affected 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 a significant hazards consideration with respect to the provisions of 10CFR50.92.