AFFECTED TECHNICAL SPECIFICATION PAGES
(NUREG-1468)

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INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

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.~~ the Core Operating Limits Report (COLR).

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 its 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 its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

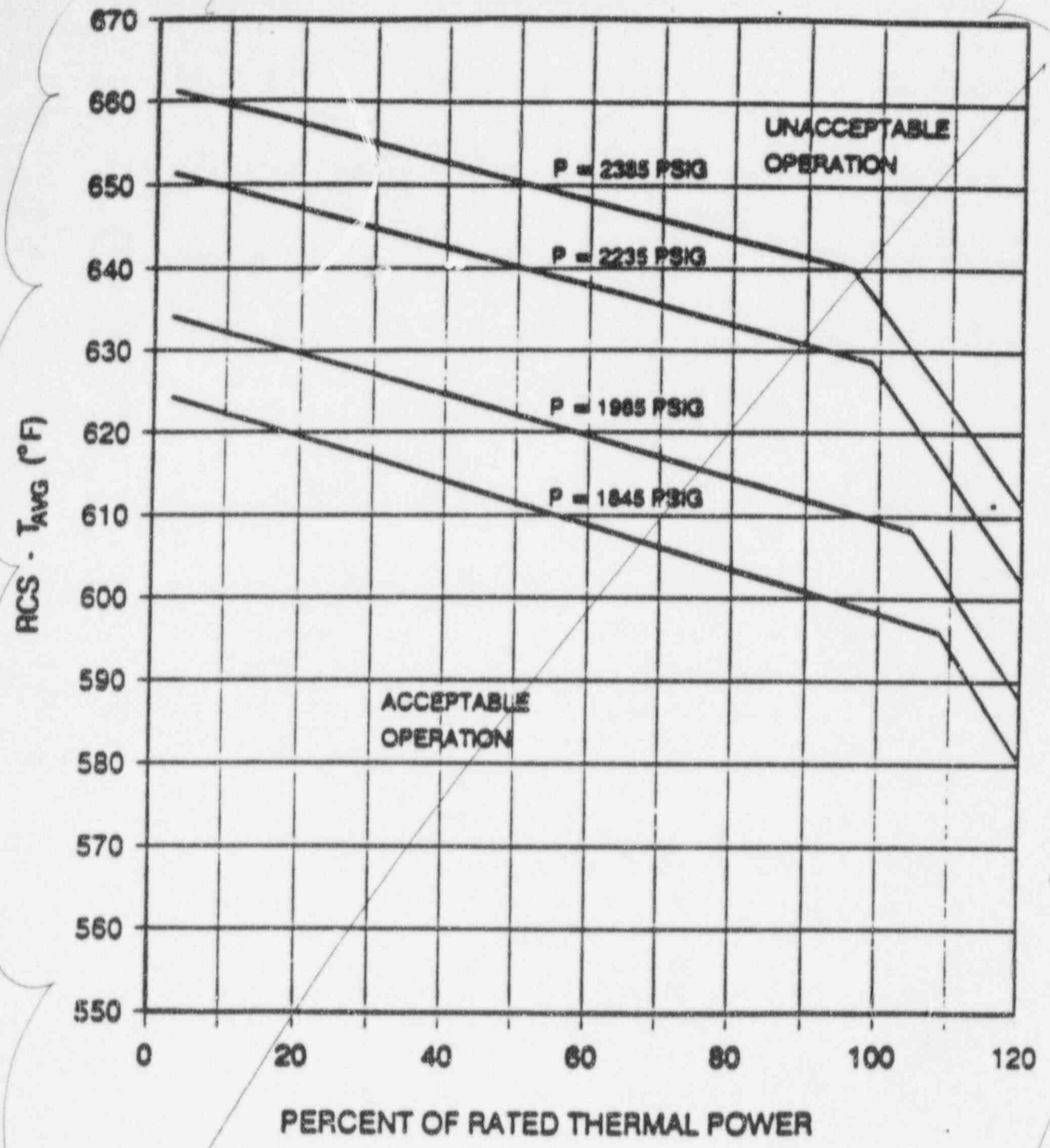

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FIGURE 2.1-1a

UNIT 1 REACTOR CORE SAFETY LIMITS

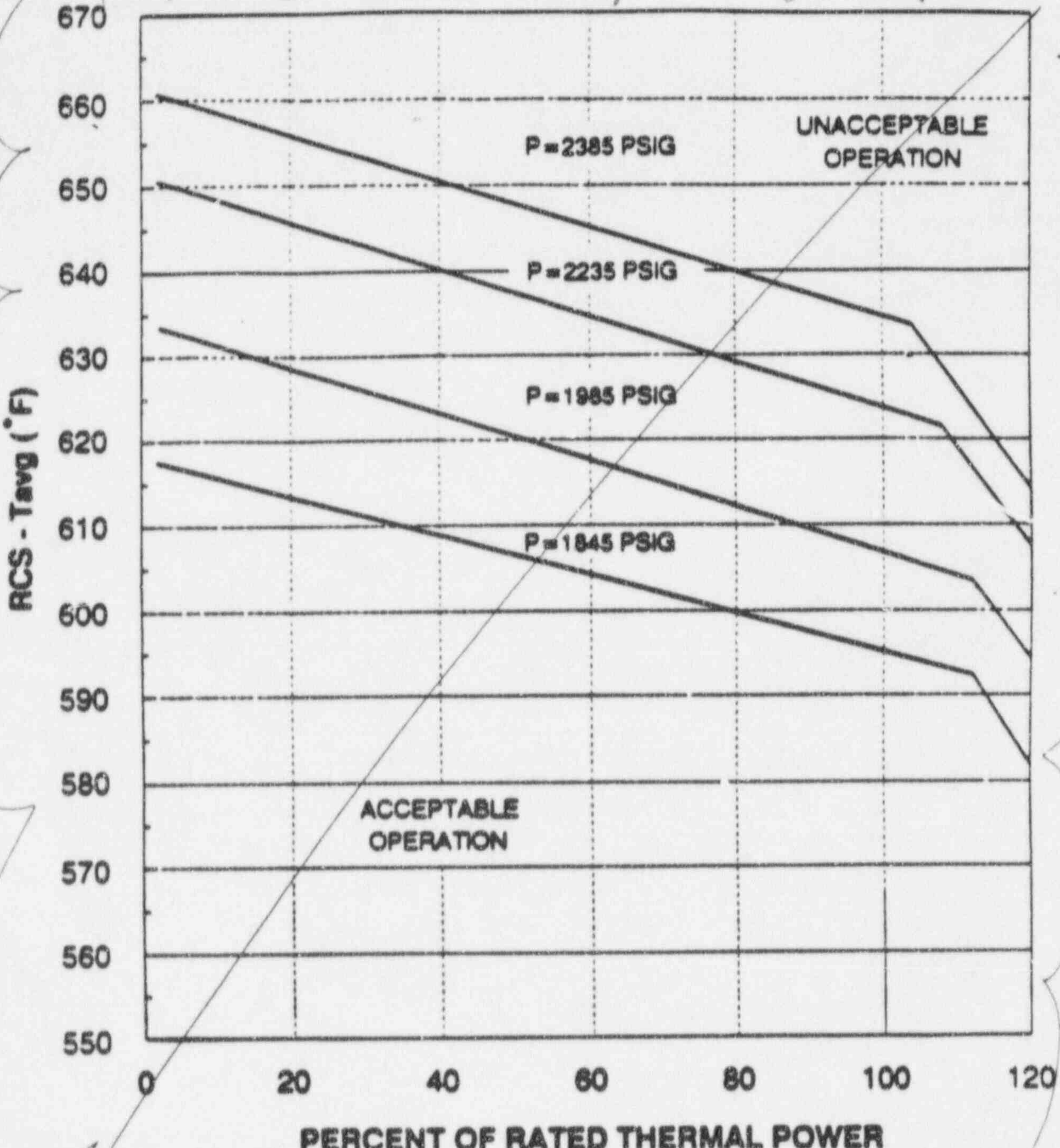

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FIGURE 2.1-1b

UNIT 2 REACTOR CORE SAFETY LIMITS

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	1.25	≤109% of RTP*	≤111.7% of RTP*
b. Low Setpoint	8.3	4.56	1.25	≤25% of RTP*	≤27.7 of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Not Used					
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤31.5 of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature N-16	**	**	**	**	**
~~a. Unit 1~~	~~10.53~~	~~6.70~~	~~1.0+1.10+0.76(1)~~	~~See Note 1~~	~~See Note 2~~
~~b. Unit 2~~	~~10.0~~	~~6.75~~	~~1.0+1.38+0.96(2)~~	~~See Note 1~~	~~See Note 2~~

*RTP = RATED THERMAL POWER

~~(1) 1.0% span for N-16 power monitor, 1.10% for T_{cold} RTDs and 0.76% for pressurizer pressure sensors.~~
~~(2) 1.0% span for N-16 power monitor, 1.38% for T_{cold} RTDs and 0.96% for pressurizer pressure sensors.~~

** As specified in the Core Operating Limits Report.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Overpower N-16	4.0	2.05	1.0+0.05(3)	≤112% of RTP*	≤114.5% of RTP*
9. Pressurizer Pressure-Low					
a. Unit 1	4.4	0.71	2.0	≥1880 psig	≥1863.6 psig
b. Unit 2	4.4	1.12	2.0	≥1880 psig	≥1863.6 psig
10. Pressurizer Pressure-High					
a. Unit 1	7.5	5.01	1.0	≤2385 psig	≤2400.8 psig
b. Unit 2	7.5	1.12	2.0	≤2385 psig	≤2401.4 psig
11. Pressurizer Water Level-High					
a. Unit 1	8.0	2.18	2.0	≤92% of instrument span	≤93.9% of instrument span
b. Unit 2	8.0	2.35	2.0	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low					
a. Unit 1	2.5	1.18	0.6	≥90% of loop design flow**	≥88.6% of loop design flow**
b. Unit 2	2.5	1.25	0.87	≥90% of loop minimum measured flow***	≥88.8% of loop minimum measured flow***

(3) 1.0% span for N-16 power monitor and 0.05% for T_{cold} RTDs.
* RTP - RATED THERMAL POWER
** Loop design flow ~~= 99,050 gpm~~ as specified in the Core Operating Limits Report (COLR)
~~*** Loop minimum measured flow - 98,500 gpm~~

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: Overtemperature N-16

$$N = K_1 - K_2 \left[\frac{1 + \tau_1 s}{1 + \tau_2 s} T_c - T_c^{o}\right] + K_3 (P - P^1) - f_1 (\Delta q)$$

Where:

	N	=	Measured N-16 Power by ion chambers,
	T_c	=	Cold leg temperature, °F,
	T_c^{o}	=	560.5°F for Unit 1, 560.3°F for Unit 2 - Reference T_c at RATED THERMAL POWER,
	K_1	=	1.150,
	K_2	=	0.0134/°F for Unit 1 0.016856/°F for Unit 2
	$\frac{1 + \tau_1 s}{1 + \tau_2 s}$	=	The function generated by the lead-lag controller for T_c dynamic compensation,
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_c, $\tau_1 \geq 10$ s, and $\tau_2 \leq 3$ s,
	K_3	=	0.000719/psig for Unit 1 0.000898/psig for Unit 2

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

P	=	Pressurizer pressure, psig,
P'	≥	2235 psig (Nominal RCS operating pressure),
S	=	Laplace transform operator, s^{-1},

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

For Unit 1

(i) for $q_t - q_b$ between -65% and +4%, $f_1(\Delta q) = 0$, 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 -65%, the N-16 Trip Setpoint shall be automatically reduced by 1.81% of its value at RATED THERMAL POWER, and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +4%, the N-16 Trip Setpoint shall be automatically reduced by 2.26% of its value at RATED THERMAL POWER.

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TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

For Unit 2

(i) for $q_t - q_b$ between -52% and +5.5%, $f_1(\Delta q) = 0$, 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 -52%, the N-16 Trip Setpoint shall be automatically reduced by 2.15% of its value at RATED THERMAL POWER, and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +5.5%, the N-16 Trip Setpoint shall be automatically reduced by 2.17% 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 3.51% of span for Unit 1 or 2.88% of span for Unit 2.

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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 heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is 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 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 level that DNB will not occur when the minimum DNBR is at the DNBR limit. In meeting this design basis, uncertainties in plant operating parameters are considered such that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

Reactor Core Safety Limits

The curves ~~of Figure 2.1-1~~ show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the safety analysis limit value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. In addition, the Source Range Neutron Flux trip provides similar protection during shutdown operations with the reactor trip breakers closed and the rod control system capable of control rod withdrawal. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature N-16

The Overtemperature N-16 trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the N-16 detectors, and pressure is within the range between the Pressurizer High and Low Pressure trips. The setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the cold leg temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the ~~core Safety Limit as shown in Figure 2.1-1~~. Reactor Core Safety Limits curves. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced. ~~according to the notations in Table 2.2-1.~~

Overpower N-16

The Overpower N-16 trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature trip, and provides a backup to the High Neutron Flux trip. The Overpower N-16 trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~1.6% Δk/k for Unit 1 (1.3% Δk/k for Unit 2)~~ the value specified in the Core Operating Limits Report (COLR).

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

ACTION:

With the SHUTDOWN MARGIN less than ~~1.6% Δk/k for Unit 1 (1.3% Δk/k for Unit 2),~~ the value specified in the COLR, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1.6% Δk/k for Unit 1 (1.3% Δk/k for Unit 2):~~ the value specified in the COLR:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to ~~1.3% Δk/k.~~ the value specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than ~~1.3% Δk/k;~~ the value specified in the COLR, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1.3% Δk/k:~~ the value specified in the COLR:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid storage tanks via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via centrifugal charging pumps to the RCS.

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

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least ~~1.3% Δk/k~~ the value specified in the COLR at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the flow path from the boric acid storage tanks is greater than or equal to 65°F when it is a required water source;

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position; and

c. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS.

*A maximum of two charging pumps shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°F except when Specification 3.4.8.3 is not applicable. An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve(s) with power removed from the valve operator(s) or by a manual isolation valve(s) secured in the closed position.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two centrifugal charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least ~~1.3% Δk/k~~ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

the value specified in the COLR

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 The required centrifugal charging pump(s) shall be demonstrated OPERABLE by testing pursuant to Specification 4.0.5.

4.1.2.4.2 The required positive displacement charging pump shall be demonstrated OPERABLE by testing pursuant to Specification 4.1.2.2.c.

4.1.2.4.3 Whenever the temperature of one or more of the Reactor Coolant System (RCS) cold legs is less than or equal to 350°F, a maximum of two charging pumps shall be OPERABLE, except when Specification 3.4.8.3 is not applicable.

When required, one charging pump shall be demonstrated inoperable# at least once per 31 days by verifying that the motor circuit breakers are secured in the open position.

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*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODES 3 and 4 for the charging pump declared inoperable pursuant to Specification 3.1.2.4 provided the charging pump is restored to OPERABLE status within 4 hours after entering MODE 3 or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

**In MODE 4 the positive displacement pump may be used in lieu of one of the required centrifugal charging pumps.

#An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve(s) with power removed from the valve operator(s) or by a manual isolation valve(s) secured in the closed position.

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:

a. A boric acid storage tank with:

1) A minimum indicated borated water level of 50%,

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 indicated borated water level of 95%,

2) A boron concentration between 2400 ppm and 2600 ppm,

3) A minimum solution temperature of 40°F, and

4) A maximum solution temperature of 120°F.

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

ACTION:

a. With the boric acid storage tank inoperable and being used as one of the above required borated water sources, restore the tank 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 ~~1.3% Δk/k~~ the value specified in the COLR at 200°F; restore the boric acid storage tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, 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.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the stated limits:

a. Indicated Reactor Coolant System $T_{avg} \leq 592°F$

b. Indicated Pressurizer Pressure ≥ 2219 psig*

c. Indicated Reactor Coolant System (RCS) Flow ** ≥ 403,400 gpm** for Unit 1
≥ 395,200 gpm** for Unit 2

the value specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the above parameters shall be verified to be within its limits at least once per 12 hours.

4.2.5.2 The RCS total flow rate shall be verified to be within its limits at least once per 31 days by plant computer indication or measurement of the RCS elbow tap differential pressure transmitters' output voltage.

4.2.5.3 The RCS loop flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The channels shall be normalized based on the RCS flow rate determination of Surveillance Requirement 4.2.5.4.

4.2.5.4 The RCS total flow rate shall be determined by precision heat balance measurement after each fuel loading and prior to operation above 85% of RATED THERMAL POWER. The feedwater pressure and temperature, the main steam pressure, and feedwater flow differential pressure instruments shall be calibrated within 90 days of performing the calorimetric flow measurement.

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**Includes a 1.8% flow measurement uncertainty.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN (as specified in the COLR) ~~of 1.6% Δk/k for Unit 1 (1.3% Δk/k for Unit 2)~~ is required to control the reactivity ~~transients~~ transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. (when Tavg is above 200°F) With T_{avg} less than 200°F, the SHUTDOWN MARGIN ~~of 1.3% Δk/k provides adequate protection and~~ is based on ~~the results of~~ (the required) the boron dilution accident analysis.

Since the actual overall core reactivity balance comparison required by 4.1.1.1.2 cannot be performed until after criticality is attained, this comparison is not required (and the provisions of Specification 4.0.4 are not applicable) for entry into any Operational Mode within the first 31 EFPD following initial fuel load or refueling.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

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

REACTIVITY CONTROL SYSTEMS

BASES

MODERATOR TEMPERATURE COEFFICIENT (Continued)

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 End of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value 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 EOL MTC value.

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 remains within its limits since 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 551°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, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

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 200°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~~ the required SHUTDOWN MARGIN from expected operating conditions ~~of 1.6% Δk/k for Unit 1 (1.3% Δk/k for Unit 2)~~ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 15,700 gallons of 7000 ppm borated water from the boric acid storage tanks or 70,702 gallons of 2400 ppm borated water from the refueling water storage tank (RWST).

REACTIVITY CONTROL SYSTEMS

BASES

BORATION SYSTEMS (Continued)

With the RCS temperature below 200°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 two charging pumps to be OPERABLE and the requirement to verify one charging pump to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The limitation for minimum solution temperature of the borated water sources are sufficient to prevent boric acid crystallization with the highest allowable boron concentration.

The boron capability required below 200°F is sufficient to provide ~~a~~ the required SHUTDOWN MARGIN ~~of 1.3% Δk/k~~ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 1,100 gallons of 7000 ppm borated water from the boric acid storage tanks or 7,113 gallons of 2400 ppm borated water from the RWST.

As listed below, the required indicated levels for the boric acid storage tanks and the RWST include allowances for required/analytical volume, unusable volume, measurement uncertainties (which include instrument error and tank tolerances, as applicable), margin, and other required volume.

Tank	MODES	Ind. Level	Unusable Volume (gal)	Required Volume (gal)	Measurement Uncertainty	Margin (gal)	Other (gal)
RWST	5,6	24%	98,900	7,113	4% of span	10,293	N/A
	1,2,3,4	95%	45,494	70,702	4% of span	N/A	357,535*
Boric	5,6	10%	3,221	1,100	6% of span	N/A	N/A
Acid	5,6	20%	3,221	1,100	6% of span	3,679	N/A
Storage	(gravity feed)						
Tank	1,2,3,4	50%	3,221	15,700	6% of span	N/A	N/A

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

*Additional volume required to meet Specification 3.5.4.

ADMINISTRATIVE CONTROLS

MONTHLY OPERATING REPORTS (Continued)

shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.6a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1). Moderator temperature coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2). Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

3). Control Rod Insertion Limits for Specification 3/4.1.3.6,

4). AXIAL FLUX DIFFERENCE Limits and target band for Specification 3/4.2.1.,

5). Heat Flux Hot Channel Factor, K(Z), W(Z), F_Q^{RTP}, and the $F_Q^{C}(Z)$ allowances for Specification 3/4.2.2,

6). Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

7) Reactor Core Safety Limits for Specification 2.1.1
8) Overtemperature N-16 Trip Setpoint and Thermal Design Flow for Specification 2.2
9) Shutdown Margin for Specifications 3/4.1.1.1, 3/4.1.1.2, 3/4.1.2.2, 3/4.1.2.4 and 3/4.1.2.6
10) Indicated Reactor Coolant System flow for Specification 3/4.2.5.

6.9.1.6b The following analytical methods used to determine the core operating limits are for Units 1 and 2, unless otherwise stated, and shall be those previously approved by the NRC in:

1). WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.),

2.1 - Reactor Core Safety Limits, 2.2 - Overtemperature N-16 Trip Setpoints and Thermal Design Flow, 3/4.1.1.1, 3/4.1.1.2, 3/4.1.2.2, 3/4.1.2.4 and 3/4.1.2.6 - Shutdown Margin, and 3/4.2.5 - Indicated Reactor Coolant System Flow.)

2). WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974 (W Proprietary). (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

3). T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

4). NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

(2.1 - Reactor Core Safety Limits, and 2.2 - Overtemperature N-16 Trip Setpoints and Thermal Design Flow.)

5). WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor (W(z) surveillance requirements for F_Q Methodology).)

6). WCAP-10079-P-A, "NOTRUMP, A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," August 1985, (W Proprietary).

7). WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE", August 1985, W Proprietary).

8). WCAP-11145-P-A, "WESTINGHOUSE SMALL BREAK LOCA ECCS EVALUATION MODEL GENERIC STUDY WITH THE NOTRUMP CODE", October 1986, W Proprietary).

9). RXE-90-006-P, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," February 1991. (Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor.) ,

10). RXE-88-102-P, "TUE-1 Departure from Nucleate Boiling Correlation", January 1989.

11). RXE-88-102-P, Sup. 1, "TUE-1 DNB Correlation - Supplement 1", December 1990.

12). RXE-89-002, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", June 1989.

(Methodology for Specification 3/4.2.5 - In-Situ Reactor Coolant System Flow.)

13). RXE-91-001, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", February 1991.

14). RXE-91-002, "Reactivity Anomaly Events Methodology", May 1991. (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

15). RXE-90-007, "Large Break Loss of Coolant Accident Analysis Methodology", December 1990.

16). TXX-88306, "Steam Generator Tube Rupture Analysis", March 15, 1988.

17). RXE-91-005, "Methodology for Reactor Core Response to Steamline Break Events," May, 1991.

(Methodology for Specifications 3/4.1.1.1, 3/4.1.1.2, 3/4.1.2.2 , 3/4.1.2.4 and 3/4.1.2.6 - Shutdown Margin.)

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

Reference 18) is for Unit 2 only:

18). WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL- 1981 Version", February 1982 (W Proprietary).

6.9.1.6c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

19). RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5," February 1994. (Methodology for Specifications 3/4.1.1.1, 3/4.1.1.2, 3/4.1.2.2, 3/4.1.2.4 and 3/4.1.2.6 - Shutdown Margin.)

ENCLOSURE 1 TO TXX-95076

GENERIC LETTER 88-16 - REMOVAL OF CYCLE-SPECIFIC PARAMETER

LIMITS FROM TECHNICAL SPECIFICATIONS

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

OCT 0 4 1988

TO ALL POWER REACTOR LICENSEES AND APPLICANTS

SUBJECT: REMOVAL OF CYCLE-SPECIFIC PARAMETER LIMITS FROM TECHNICAL SPECIFICATIONS (GENERIC LETTER 88-16)

License amendments are generally required each fuel cycle to update the values of cycle-specific parameter limits in Technical Specifications (TS). The processing of changes to TS that are developed using an NRC-approved methodology is an unnecessary burden on licensee and NRC resources. A lead-plant proposal for an alternative that eliminates the need for a license amendment to update the cycle-specific parameter limits each fuel cycle was submitted for the Oconee plant with the endorsement of the Babcock and Wilcox Owners Group. On the basis of the NRC review and approval of that proposal, the enclosed guidance for the preparation of a license amendment request for this alternative was developed by the NRC staff.

Generally, the methodology for determining cycle-specific parameter limits is documented in an NRC-approved Topical Report or in a plant-specific submittal. As a consequence, the NRC review of proposed changes to TS for these limits is primarily limited to confirmation that the updated limits are calculated using an NRC-approved methodology and consistent with all applicable limits of the safety analysis. These changes also allow the NRC staff to trend the values of these limits relative to past experience. This alternative allows continued trending of these limits without the necessity of prior NRC review and approval.

Licensees and applicants are encouraged to propose changes to TS that are consistent with the guidance provided in the enclosure. Conforming amendments will be expeditiously reviewed by the NRC Project Manager for the facility. Proposed amendments that deviate from this guidance will require a longer, more detailed review. Please contact the Project Manager if you have questions on this matter.

Sincerely,

8810050058

Dennis M. Crutchfield

Dennis M. Crutchfield
Acting Associate Director for Projects
Office of Nuclear Reactor Regulation

RECEIVED

OCT 2 1 1988

WILLIAM G. COUNSIL

Enclosure:
As stated

GUIDANCE FOR TECHNICAL SPECIFICATION CHANGES FOR CYCLE-SPECIFIC PARAMETER LIMITS

INTRODUCTION

A number of Technical Specifications (TS) address limits associated with reactor physics parameters that generally change with each reload core, requiring the processing of changes to TS to update these limits each fuel cycle. If these limits are developed using an NRC-approved methodology, the license amendment process is an unnecessary burden on the licensee and the NRC. An alternative to including the values of these cycle-specific parameters in individual specifications is provided and is responsive to industry and NRC efforts on improvements in TS.

This enclosure provides guidance for the preparation of a license amendment request to modify TS that have cycle-specific parameter limits. An acceptable alternative to specifying the values of cycle-specific parameter limits in TS was developed on the basis of the review and approval of a lead-plant proposal for this change to the TS for the Oconee units. The implementation of this alternative will result in a resource savings for the licensees and the NRC by eliminating the majority of license amendment requests on changes in values of cycle-specific parameters in TS.

DISCUSSION

This alternative consists of three separate actions to modify the plant's TS: (1) the addition of the definition of a named formal report that includes the values of cycle-specific parameter limits that have been established using an NRC-approved methodology and consistent with all applicable limits of the safety analysis, (2) the addition of an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the Commission for information, and (3) the modification of individual TS to note that cycle-specific parameters shall be maintained within the limits provided in the defined formal report.

In the evaluation of this alternative, the NRC staff concluded that it is essential to safety that the plant is operated within the bounds of cycle-specific parameter limits and that a requirement to maintain the plant within the appropriate bounds must be retained in the TS. However, the specific values of these limits may be modified by licensees, without affecting nuclear safety, provided that these changes are determined using an NRC-approved methodology and consistent with all applicable limits of the plant safety analysis that are addressed in the Final Safety Analysis Report (FSAR). Additionally, it was concluded that a formal report should be submitted to NRC with the values of these limits. This will allow continued trending of this information, even though prior NRC approval of the changes to these limits would not be required.

The current method of controlling reactor physics parameters to assure conformance to 10 CFR 50.36 is to specify the specific value(s) determined to be within specified acceptance criteria (usually the limits of the safety analyses) using an approved calculation methodology. The alternative contained in this guidance controls the values of cycle-specific parameters and assures conformance to 10 CFR 50.36, which calls for specifying the lowest functional

performance levels acceptable for continued safe operation, by specifying the calculation methodology and acceptance criteria. This permits operation at any specific value determined by the licensee, using the specified methodology, to be within the acceptance criteria. The Core Operating Limits Report will document the specific values of parameter limits resulting from licensee's calculations including any mid-cycle revisions to such parameter values.

The following items show the changes to the TS for this alternative. A defined formal report, "Core Operating Limits Report" (the name used as an example for the title for this report), shall be added to the Definitions section of the TS, as follows.

> <u>[CORE] OPERATING LIMITS REPORT</u>
> 1.XX The [CORE] OPERATING LIMITS REPORT is the unit-specific document that provides [core] operating limits for the current operating reload cycle. These cycle-specific [core] operating limits shall be determined for each reload cycle in accordance with Specification 6.9.X. Plant operation within these operating limits is addressed in individual specifications.

A new administrative reporting requirement shall be added to existing reporting requirements, as follows.

> <u>[CORE] OPERATING LIMITS REPORT</u>
> [6.9.X] [Core] operating limits shall be established and documented in the [CORE] OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. (If desired, the individual specifications that address [core] operating limits may be referenced.) The analytical methods used to determine the [core] operating limits shall be those previously reviewed and approved by NRC in [identify the Topical Report(s) by number, title, and date, or identify the staff's safety evaluation report for a plant-specific methodology by NRC letter and date]. The [core] operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The [CORE] OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Individual specifications shall be revised to state that the values of cycle-specific parameters shall be maintained within the limits identified in the defined formal report. Typical modifications for individual specifications are as follows.

> The regulating rods shall be positioned <u>within the acceptable operating range for regulating rod position provided in the [CORE] OPERATING LIMITS REPORT</u>. (Used where the operating limit covers a range of acceptable operation, typically defined by a curve.)

> The [cycle-specific parameter] shall be <u>within the limit provided in the [CORE] OPERATING LIMITS REPORT.</u> (Used where the operating limit has a single point value.)

SUMMARY

The alternative to including the values of cycle-specific parameter limits in individual specifications includes (1) the addition of a new defined term for the formal report that provides the cycle-specific parameter limits, (2) the addition of its associated reporting requirement to the Administrative Controls section of the TS, and (3) the modification of individual specifications to replace these limits with a reference to the defined formal report for the values of these limits. With this alternative, reload license amendments for the sole purpose of updating cycle-specific parameter limits will be unnecessary.