Docket No. 50-423
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Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Proposed Technical Specification Changes

Cycle 4

November 1990

Millstone Nuclear Power Station, Unit No. 3
Proposed Technical Specification Changes, Cycle 4

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DEFINITIONS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

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VENTING

1.39 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

SPENT FUEL POOL STORAGE PATTERNS:

1.40 Region I spent fuel racks contain a cell blocking device in every 4th location for criticality control. This 4th location will be referred to as the blocked location. A STORAGE PATTERN refers to the blocked location and all adjacent and diagonal Region I cell locations surrounding the blocked location. Boundary configuration between Region I and Region II must have cell blockers positioned in the outermost row of the Region I perimeter, as shown in Figure 3.9-2.

1.41 Region II contains no cell blockers.

CORE OPERATING LIMITS REPORT (COLR)

1.42 The CORE OPERATING LIMITS REPORT (COLR) 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 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

ALLOWED POWER LEVEL

1.43 APL^{ND} is the minimum allowable nuclear design power level for base load operation and is specified in the COLR.

1.44 APL^{BL} is the maximum allowable power level when transitioning into base load operation.

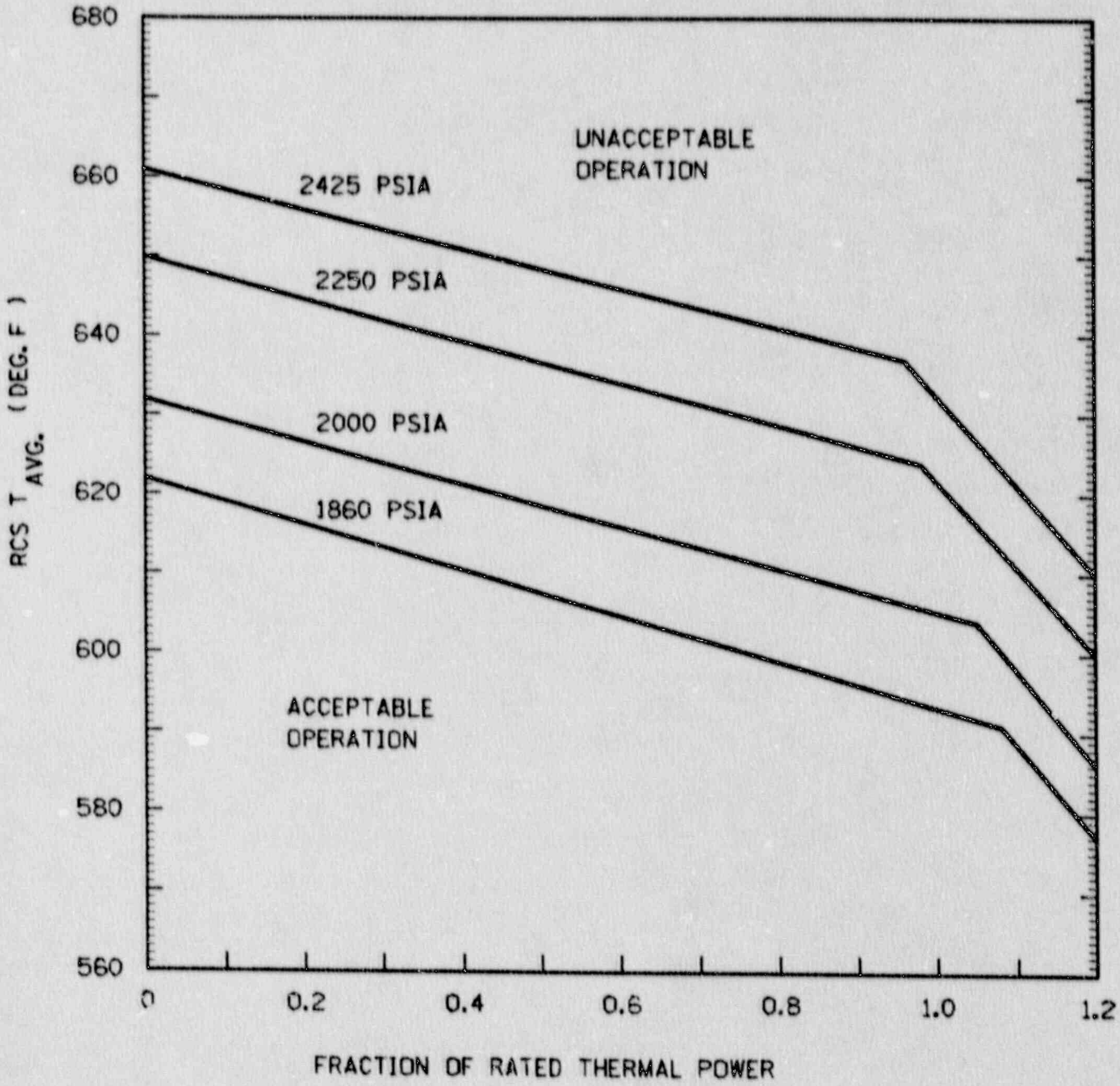

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

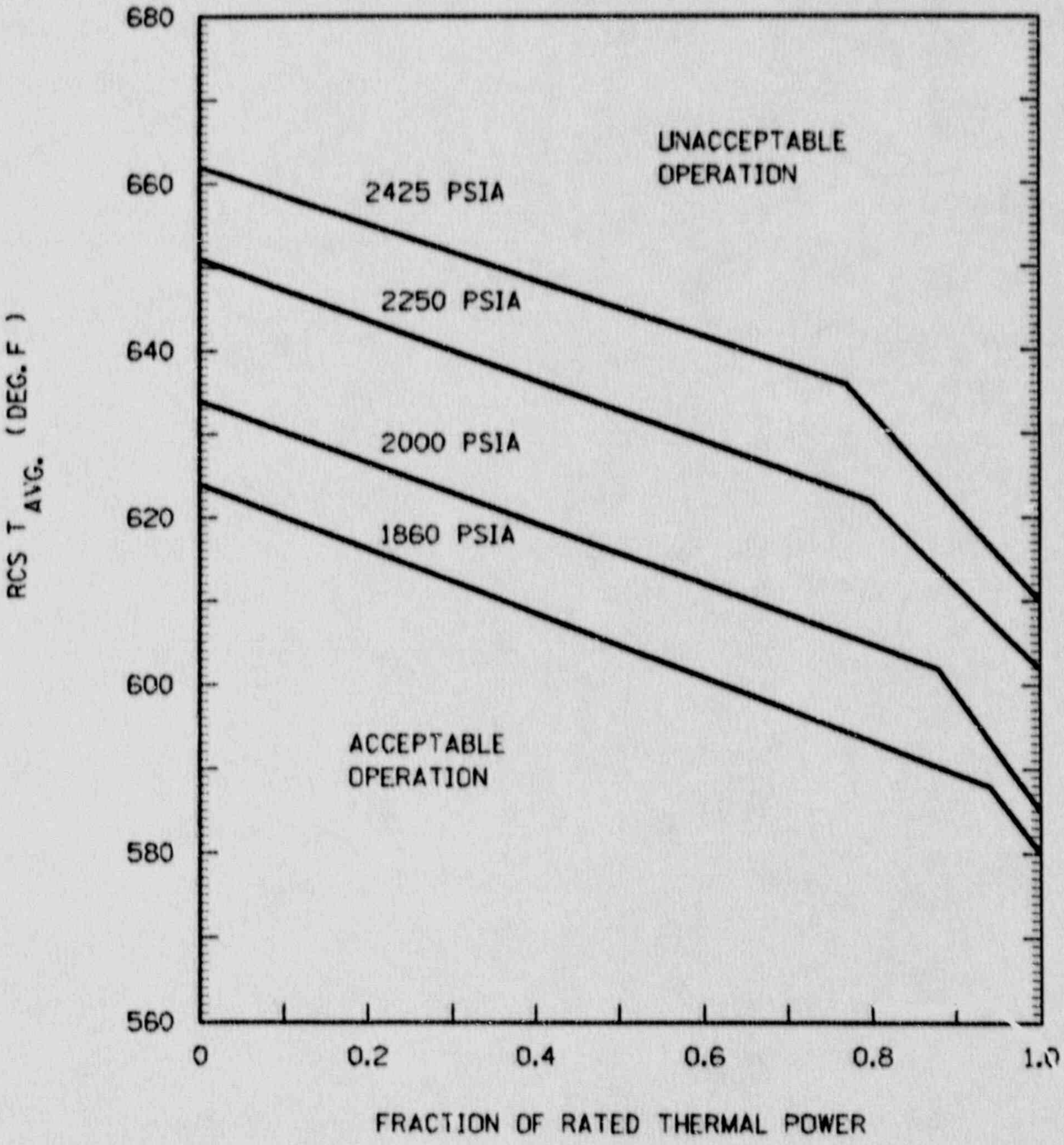

FIGURE 2.1-2

REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

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					
1) Four Loops Operating	7.5	4.56	0	≤ 109% of RTP**	≤ 111.1% of RTP**
2) Three Loops Operating	7.5	4.56	0	≤ 80% of RTP**	≤ 82.1% of RTP**
b. Low Setpoint	8.3	4.56	0	≤ 25% of RTP**	≤ 27.1% 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. Power Range, Neutron Flux, High Negative 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
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% 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 ΔT					
a. Four Loops Operating					
1) Channels I, II	10.0	6.80	1.71 + 1.33 (Temp + Press)	See Note 1	See Note 2
2) Channels III, IV	10.0	5.83	1.71 + 2.60 (Temp + Press)	See Note 1	See Note 2

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
b. Three Loops Operating					
1) Channels I, II	10.0	6.80	1.71 + 1.33 (Temp + Press)	See Note 1	See Note 2
2) Channels III, IV	10.0	5.83	1.71 + 2.60 (Temp + Press)	See Note 1	See Note 2
8. Overpower ΔT	4.8	1.24	1.71	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	1.77	3.3	≥ 1900 psia	≥ 1890 psia
10. Pressurizer Pressure-High	5.0	1.77	3.3	≤ 2385 psia	≤ 2395 psia
11. Pressurizer Water Level-High	8.0	5.13	2.7	≤ 89% of instrument span	≤ 90.7% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.52	0.78	≥ 90% of loop design flow*	≥ 89.1% of loop design flow*
13. Steam Generator Water Level Low-Low	18.10	16.64	1.50	≥ 18.10% of narrow range instrument span	≥ 17.11% of narrow range instrument span
14. General Warning Alarm	N.A.	N.A.	N.A.	N.A.	N.A.
15. Low Shaft Speed - Reactor Coolant Pumps	3.8	0.5	0	≥ 95.8% of rated speed	≥ 92.5% of rated speed

*Minimum Measured Flow Per Loop = 96,870 gpm (Four Loops Operating); 101,066 gpm (Three Loops Operating)

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	$\geq$ 500 psig	$\geq$ 450 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	$\geq$ 1% open	$\geq$ 1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	$\leq$ 10% of RTP**	$\leq$ 12.1% of RTP**
2) P-13 input	N.A.	N.A.	N.A.	$\leq$ 10% RTP** Turbine Impulse Pressure Equivalent	$\leq$ 12.1% RTP** Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8					
1) Four Loops Operating	N.A.	N.A.	N.A.	$\leq$ 37.5% of RTP**	$\leq$ 39.6% of RTP**
2) Three Loops Operating	N.A.	N.A.	N.A.	$\leq$ 37.5% of RTP**	$\leq$ 39.6% of RTP**

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	≤ 51% of RTP**	≤ 53.1% of RTP**
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A	≥ 10% of RTP**	≥ 7.9% of RTP**
19. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.
21. Three Loop Operation Bypass Circuitry	N.A.	N.A.	N.A.	N.A.	N.A

**RTP = RATED THERMAL POWER

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 Reactor Coolant System 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_0 = Indicated ΔT at RATED THERMAL POWER;

K_1 = 1.20 (Four Loops Operating); 1.20 (Three Loops Operating);

K_2 = 0.02456;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = 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 = 20 s, τ_5 = 4 s;

T = Average temperature, °F;

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

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

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	587.1°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.001311/psi;
P	=	Pressurizer pressure, psia;
P'	=	2250 psia (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:

(1) For $q_t - q_b$ between -26% and + 3%, $f_1(\Delta I) = 0$, 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;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -26%, the ΔT Trip Setpoint shall be automatically reduced by 3.55% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds +3%, the ΔT Trip Setpoint shall be automatically reduced by 1.98% 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 2.7% ΔT span (Four Loop Operation); 2.7% ΔT span (Three Loop Operation).

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER $\Delta^{o}T$

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{(1 + \tau_3 S)}\right) \leq \Delta T_0 \{K_4 - K_5 \left(\frac{\tau_7 S}{(1 + \tau_7 S)}\right) \left(\frac{1}{(1 + \tau_6 S)}\right) T - K_6 [T \left(\frac{1}{(1 + \tau_6 S)}\right) - T''] - f_2 (\Delta I)\}$$

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 = 1.09,

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.00180/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

T = As defined in Note 1,

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 587.1°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 2.7% ΔT span.

2.1.1 REACTOR CORE

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

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

The DNB design basis is as follows: uncertainties in the WRB-1 or WRB-2 correlations, plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a 95 percent probability with 95 percent confidence level that DNBR will not occur on the most limiting fuel rod during Condition I and II events. This establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

The curves of Figures 2.1-1 and 2.1-2 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 design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.70 (includes measurement uncertainty) and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.70\ [1 + 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming axial imbalance is within the limits of F_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

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. 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. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Trip System.

Overtemperature ΔT

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

Operation with a reactor coolant loop out of service requires Reactor Trip System modification. Three loop operation is permissible after resetting the K1 input to the Overtemperature ΔT channels, reducing the Power Range Neutron Flux High setpoint to a value just above the three loop maximum permissible power level, and resetting the P-8 setpoint to its three loop value. These modifications have been chosen so that, in three loop operation, each component of the Reactor Trip System performs its normal four loop function, prevents operation outside the safety limit curves, and prevents the DNBR from going below the design limit during normal operational and anticipated transients.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Low Shaft Speed - Reactor Coolant Pumps

The Low Shaft Speed - Reactor Coolant Pumps trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant pump speed (with resulting decrease in flow) on two reactor coolant pumps in any two operating reactor coolant loops. The trip setpoint ensures that a reactor trip will be generated, considering instrument errors and response times, in sufficient time to allow the DNBR to be maintained greater than the design limit following a four-pump loss of flow event.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by P-9 (a power level of approximately 50% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip) and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump low shaft speed, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% $\Delta k/k$ for both four loop and three loop operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the SHUTDOWN MARGIN less than 1.3% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6300 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 1.3% $\Delta k/k$:

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 4.1.1.1.2, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least 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.

The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

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 volume of 6700 gallons,

2) A boron concentration between 6300 and 7175 ppm, and

3) A minimum solution temperature of 67°F.

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

1) A minimum contained borated water volume of 250,000 gallons,

2) A minimum boron concentration of 2700 ppm, and

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

3) Verifying the Boric Acid Transfer Pump Room temperature and 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.

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 System with:

1) A minimum contained borated water volume of 23,620 gallons,

2) A boron concentration between 6300 and 7175 ppm, and

3) A minimum solution temperature of 67°F.

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

1) A minimum contained borated water volume of 1,166,000 gallons,

2) A boron concentration between 2700 and 2900 ppm,

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

4) A maximum solution temperature of 50°F.

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

ACTION:

a. With the Boric Acid Storage System inoperable, 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 1.3% $\Delta k/k$ 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, 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.

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 65% of RATED THERMAL POWER with the reactor coolant stop valves in the nonoperating loop closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.1.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. The limits specified in the CORE OPERATING LIMITS REPORT (COLR) for Relaxed Axial Offset Control (RAOC) operation, or

b. Within the target band about the target flux difference during base load operation, specified in the COLR.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*.

ACTION:

a. For RAOC operation with the indicated AFD outside of the applicable limits specified in the COLR,

   1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux--High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. For base load operation above APL^{ND} with the indicted AFD outside of the applicable target band about the target flux differences:

   1. Either restore the indicated AFD to within the COLR specified target band within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue base load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

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*See Special Test Exception 3.10.2

SURVEILLANCE REQUIREMENTS

4.2.1.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at least once per 7 days when the AFD Monitor Alarm is OPERABLE:

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.1.3 When in base load operation, the target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.1.4 When in base load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.1.3 or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

AXIAL FLUX DIFFERENCE

THREE LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.1.2 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. The limits specified in the CORE OPERATING LIMITS REPORT (COLR) for Relaxed Axial Offset Control (RAOC) operation, or

b. Within the target band specified in the COLR about the target flux difference during base load operation.

APPLICABILITY: MODE 1 above 37.5% of RATED THERMAL POWER.*

ACTION:

a. For RAOC operation with the indicated AFD outside of the applicable limits specified in the COLR,

   1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 37.5% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux--High Trip setpoints to less than or equal to 41% of RATED THERMAL POWER within the next 4 hours.

b. For base load operation above APL^{ND} with the indicated AFD outside of the applicable target band about the target flux differences:

   1. Either restore the indicated AFD to within the COLR specified target band within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue base load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 37.5% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS

4.2.1.2.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 37.5% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at least once per 7 days when the AFD Monitor Alarm is OPERABLE:

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.2.3 When in base load operation, the target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.2.4 When in base load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.2.3 or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.2.1 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5$$

F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) provided in the core operating limits report (COLR).

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit, and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2.1 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(z) \le \frac{F_Q^{RTP} \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \le \frac{F_Q^{RTP} \times K(z)}{W(z) \times 0.5} \text{ for } P \le 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle-dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z), and W(z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.6.

d. Measuring $F_Q^M(z)$ according to the following schedule:

(1) Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

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*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and power distribution map outlined.

(2) At least once per 31 Effective Full Power Days, whichever occurs first.

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) increasing since the previous determination of $F_Q^M(z)$, either of the following actions shall be taken:

(1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.1.2c, or

(2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationships specified in Specification 4.2.2.1.2c not being satisfied:

(1) Calculate the maximum percent over the core height (z) that $F_Q(z)$ exceeds its limit by the following expression:

$$\left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{P} \times K(z)}\right] - 1 \quad \times 100 \text{ for } P \geq 0.5$$

$$\left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{0.5} \times K(z)} - 1 \right] \times 100 \text{ for } P < 0.5$$

(2) One of the following actions shall be taken:

(a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the applicable AFD limits by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.1.2f.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

(b) Comply with the requirements of Specification 3.2.2.1 for $F_Q(z)$ exceeding its limit by the percent calculated, or

(c) Verify that the requirements of Specification 4.2.2.1.3 for base load operation are satisfied and enter base load operation.

g. The limits specified in Specifications 4.2.2.1.2c, 4.2.2.1.2e, and 4.2.2.1.2f above are not applicable in the following core plane regions:

(1) Lower core region from 0% to 15%, inclusive.
(2) Upper core region from 85% to 100%, inclusive.

4.2.2.1.3 Base load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering base load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.1.2 for at least the previous 24 hours. Maintain base load operation surveillance (AFD within the target band limit the target flux difference of Specification 3.2.1.1) during this time period. Base load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and

100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.1.4. APL^{BL} is defined as the minimum value of:

$$APL^{BL} = \frac{F_Q^{RTP} \times K(z)}{F_Q^M(z) \times W(z)_{BL}} \times 100\%$$

over the core height (z) where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is F_Q^{RTP}. $W(z)_{BL}$ is the cycle-dependent function that accounts for limited power distribution transient encountered during base load operation. F_Q^{RTP}, K(z), and $W(z)_{BL}$ are specified in the COLR as per Specification 6.9.1.6.

b. During base load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.1.3.a shall be satisfied before reentering base load operation.

4.2.2.1.4 During base load operation $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2.1 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$. F_Q^{RTP} is the F_Q limit, the normalized $F_Q(z)$ as a function of core height. P is the relative THERMAL POWER. $W(z)_{BL}$ is the cycle-dependent function that accounts

for limited power distribution transients encountered during base load operation. F_Q^{RTP}, $K(z)$, and $W(z)_{BL}$ are specified in the COLR as per Specification 6.9.1.6.

d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:

(1) Prior to entering base load operation after satisfying Section 4.2.2.1.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and

(2) At least once per 31 Effective Full Power Days.

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) increasing since the previous determination of $F_Q^M(z)$, either of the following actions shall be taken:

(1) $F_Q^M(z)$ shall be increased by 2% over that specified in 4.2.2.1.4.c, or

(2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until 2 successive maps indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationship specified in 4.2.2.1.4.c not being satisfied, either of the following actions shall be taken:

(1) Place core in an equilibrium condition where the limit in 4.2.2.1.2.C is satisfied, and remeasure $F_Q^M(z)$, or

(2) Comply with the requirements of Specification 3.2.2.1 for $F_Q(z)$ exceeding its limit by the maximum percent calculated over the core height (z) with the following expression:

$$\left[\frac{F_Q^M(z) \times W(z)_{BL}}{\frac{F_Q^{RTP}}{P} \times K(z)} - 1 \right] \times 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.1.4.c, 4.2.2.1.4.e, and 4.2.2.1.4.f are not applicable in the following core plane regions:

(1) Lower core region 0% to 15%, inclusive.
(2) Upper core region 85% to 100%, inclusive.

4.2.2.1.5 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.1.2, an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

THREE LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} [K(Z)] \text{ for } P > 0.375$$

$$F_Q(Z) \leq \left(\frac{F_Q^{RTP}}{0.375}\right) [K(Z)] \text{ for } P \leq 0.375$$

F_Q^{RTP} = The F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR).

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a. above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2.2 are satisfied.

c. Satisfy the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)} \text{ for } P > 0.375$$

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{W(z) \times 0.375} \text{ for } P \leq 0.375$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, $K(z)$ is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and $W(z)$ is the cycle-dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, $K(z)$, and $W(z)$ are specified in the COLR as per Specification 6.9.1.6.

d. Measuring $F_Q^M(z)$ according to the following schedule:

(1) Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined, * or

---

*During power escalation at the beginning of each cycle, the power level may be increased until a power level for extended operation has been achieved and power distribution map obtained.

(2) At least once per 31 Effective Full Power Days, whichever occurs first.

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) increasing since the previous determination of $F_Q^M(z)$, either of the following actions shall be taken:

(1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2.2c, or

(2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationships specified in Specification 4.2.2.2.2c not being satisfied:

(1) Calculate the maximum percent over the core height (z) that $F_Q(z)$ exceeds its limit by the following expression:

$$\left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{P} \times K(z)} - 1\right] \times 100 \text{ for } P \geq 0.375$$

$$\left[\frac{F_Q^M(z) \times W(z)}{\dfrac{F_Q^{RTP}}{0.375} \times K(z)}\right] - 1 \quad \times 100 \text{ for } P < 0.375$$

(2) One of the following actions shall be taken:

(a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the applicable AFD limits by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2.2f.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

(b) Comply with the requirements of Specification 3.2.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated, or

(c) Verify that the requirements of Specification 4.2.2.2.3 for base load operation are satisfied and enter base load operation.

g. The limits specified in Specifications 4.2.2.2.2c, 4.2.2.2.2e, and 4.2.2.2.2f are not applicable in the following core plane regions:

(1) Lower core region from 0% to 15%, inclusive.
(2) Upper core region from 85% to 100%, inclusive.

4.2.2.2.3 Base load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering base load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2.2 for at least the previous 24 hours. Maintain base load operation surveillance (AFD within the target band limit about the target flux difference of Specification 3.2.1.2) during this time period.

Base load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q^{BL} surveillance is maintained pursuant to Specification 4.2.2.2.4. APL^{BL} is defined as the minimum value of:

$$APL^{BL} = \frac{F_Q^{RTP} \times K(z)}{F_Q^M(z) \times W(z)_{BL}} \times 100\%$$

over the core height (z) where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is F_Q^{RTP}. $W(z)_{BL}$ is the cycle-dependent function that accounts for limited power distribution transient encountered during base load operation. F_Q^{RTP}, K(z), and $W(z)_{BL}$ are specified in the COLR as per Specification 6.9.1.6.

b. During base load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.2.3.a shall be satisfied before reentering base load operation.

4.2.2.2.4 During base load operation $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2.2 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$. The F_Q^{RTP} is the F_Q limit, the normalized $F_Q(z)$ as a function of core height. P is the relative THERMAL POWER. $W(z)_{BL}$ is the cycle-dependent function that accounts for limited power distribution transients encountered during base load operation. F_Q^{RTP}, K(z), and $W(z)_{BL}$ are specified in the COLR as per Specification 6.9.1.6.

d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:

(1) Prior to entering base load operation after satisfying Section 4.2.2.2.3, unless a full core flux map has been taken in the previous 31 Effective Full Power Days with the relative THERMAL POWER having been maintained above APL^{ND} for the 24 hours prior to mapping, and

(2) At least once per 31 Effective Full Power Days.

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) increasing since the previous determination of $F_Q^M(z)$, either of the following actions shall be taken:

(1) $F_Q^M(z)$ shall be increased by 2 percent over that specified in 4.2.2.2.4.c, or

(2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until 2 successive maps indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationship specified in 4.2.2.2.4.c not being satisfied, either of the following actions shall be taken:

(1) Place core in an equilibrium condition where the limit in 4.2.2.2.2.c is satisfied, and remeasure $F_Q^M(z)$, or

(2) Comply with the requirements of Specification 3.2.2.2 for $F_Q(z)$ exceeding its limit by the maximum percent calculated over the core height (z) with the following expression:

$$\left[\frac{F_Q^M(z) \times W(z)_{BL}}{\frac{2.25}{P} \times K(z)} - 1\right] \times 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.2.4.c, 4.2.2.2.4.e, and 4.2.2.4.f are not applicable in the following core plane regions:

(1) Lower core region 0% to 15%, inclusive.
(2) Upper core region 85% to 100%, inclusive.

4.2.2.2.5 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2.2, an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.3.1 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate $\geq$ 387,480 gpm, and

b. $F^N_{\Delta H} \leq F^{RTP}_{\Delta H} [1.0 + PF_{\Delta H} (1.0 - P)]$

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.1b. takes into consideration a measurement uncertainty of 4% for incore measurement,

3) $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER in the CORE OPERATING LIMITS REPORT (COLR),

4) $PF_{\Delta H}$ - The power factor multiplier for $F^N_{\Delta H}$ provided in the COLR, and

5) The measured value of RCS total flow rate shall be used since uncertainties of 2.4% for flow measurement have been included in Specification 3.2.3.1a.

APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours either:

1. Restore the 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.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that $F^N_{\Delta H}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.1.2 RCS total flow rate and $F^N_{\Delta H}$ shall be determined to be within the acceptable range:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.1.3 The indicated RCS total flow rate shall be verified to be within the acceptable range at least once per 12 hours when the most recently obtained value of $F^N_{\Delta H}$, obtained per Specification 4.2.3.1.2, is assumed to exist.

4.2.3.1.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.1.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.3.1.6 If the feedwater venturis are not inspected at least once per 18 months, an additional 0.1% will be added to the total RCS flow measurement uncertainty.

RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

THREE LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.3.2 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate $\geq$ 303,200 gpm, and

b. $F^N_{\Delta H} \leq F^{RTP}_{\Delta H} [1.0 + PF_{\Delta H} (1.0 - P)]$

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map.

The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.2b. takes into consideration a measurement uncertainty of 4% for incore measurement,

3) $F^{RTP}_{\Delta H}$ = The $F_{\Delta H}$ limit at RATED THERMAL POWER in the CORE OPERATING LIMITS REPORT (COLR),

4) $PF_{\Delta H}$ = The power factor multiplier for $F_{\Delta H}$ in the COLR, and

5) The measured value of RCS total flow rate shall be used since uncertainties of 2.8% for flow measurement have been included in Specification 3.2.3.2a.

APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours either:

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

2. Reduce THERMAL POWER to less than 32% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 37% of RATED THERMAL POWER within the next 4 hours.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that $F^N_{\Delta H}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 32% of RATED THERMAL POWER, and

2. A nominal 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2.2 RCS total flow rate and $F^N_{\Delta H}$ shall be determined to be within the acceptable range at least once per 31 Effective Full Power Days.

4.2.3.2.3 The indicated RCS total flow rate shall be verified to be within the acceptable range at least once per 12 hours when the most recently obtained value of $F^N_{\Delta H}$, obtained per Specification 4.2.3.2.2, is assumed to exist.

4.2.3.2.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

4.2.3.2.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.3.2.6 If the feedwater venturis are not inspected at least once per 18 months, an additional 0.1% will be added to the total RCS flow measurement uncertainty.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Within 2 hours either:

      a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exceptions Specification 3.10.2.

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

ACTION (Continued)

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

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

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 Each of the parameters of Table 3.2-1 shall be verified to be within its limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: Four Loops in Operation	LIMITS: Three Loops in Operation & Loop Stop Valves Closed
Indicated Reactor Coolant System T_{avg}	$\leq$ 591.1°F	$\leq$ 583.3°F
Indicated Pressurizer Pressure	$\geq$ 2218 psia*	$\geq$ 2218 psia*

*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.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	≤ 7 seconds*
8. Overpower ΔT	≤ 7 seconds*
9. Pressurizer Pressure--Low	≤ 2 seconds
10. Pressurizer Pressure--High	≤ 2 seconds
11. Pressurizer Water Level--High	≤ 2 seconds

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE, with at least three reactor coolant loops in operation when the Reactor Trip System breakers are closed or with at least one reactor coolant loop in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than the above required reactor coolant loops in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 17% at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated with power removed from the associated RCS loop stop valve operators until:

a. The temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops,

b. The boron concentration of the isolated loop is greater than or equal to the boron concentration of the operating loops, or greater than 2600 ppm whichever is less,

c. The isolated portion of the loop has been drained and is refilled, and

d. The reactor is subcritical by at least 1.6% $\Delta k/k$.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the requirements of the above specification not satisfied, do not open the isolated loop stop valves.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.2 The reactor shall be determined to be subcritical by at least 1.6% $\Delta k/k$ within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.3 Within 4 hours prior to opening the loop stop valves, the isolated loop shall be determined to:

a. Be drained and refilled, and

b. Have a boron concentration greater than or equal to the boron concentration of the operating loops, or greater than 2600 ppm whichever is less.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6618 and 7030 gallons,

c. A boron concentration of between 2600 and 2900 ppm, and

d. A nitrogen cover-pressure of between 636 and 694 psia.

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 8 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig 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 reduce pressurizer pressure to less than 1000 psig 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 volume and nitrogen cover-pressure in the tanks to be within the above limits, and

2) Verifying that each accumulator isolation valve is open.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution; and

---

*Pressurizer pressure above 1000 psig.

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection actuation test signal, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

3) Verifying that the Residual Heat Removal pumps stop automatically upon receipt of a Low-Low RWST Level test signal.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump $\geq$ 2411 psid,

2) Safety Injection pump $\geq$ 1348 psid,

3) RHR pump $\geq$ 165 psid, and

4) Containment recirculation pump $\geq$ 130 psid.

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

ECCS Throttle Valves

Valve Number	Valve Number
3SIH*V6	3SIH*V25
3SIH*V7	3SIH*V27

ECCS Throttle Valves Valve Number	Valve Number
3SIH*V8	3SIH*V107
3SIH*V9	3SIH*V108
3SIH*V21	3SIH*V109
3SIH*V23	3SIH*V111

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 339 gpm, and

b) The total pump flow rate is less than or equal to 560 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 442.5 gpm, and

b) The total pump flow rate is less than or equal to 670 gpm for the A pump and 650 gpm for the B pump.

3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3976 gpm.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE witn:

a. A contained borated water volume between 1,166,000 and 1,207,000 gallons,

b. A boron concentration between 2700 and 2900 ppm of boron,

c. A minimum solution temperature of 40°F, and

d. A maximum solution temperature of 50°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

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 The Spray Additive System shall be OPERABLE with:

a. A chemical addition tank containing a volume of between 17,760 and 18,760 gallons of between 3.4 and 4.1% by weight NaOH solution, and

b. Two gravity feed paths each capable of adding NaOH solution from the chemical addition tank to each Containment Quench Spray subsystem pump suction.

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

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The Spray Additive System shall be demonstrated OPERABLE:

a. 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;

b. At least once per 6 months by:

1) Verifying the contained solution volume in the tank, and

2) Verifying the concentration of the NaOH solution by chemical analysis is within the above limits.

c. At least once per 18 months, during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.1 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; either:

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

b. A boron concentration of greater than or equal to 2600 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 33 gpm of a solution containing greater than or equal to 6300 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 2600 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1.1 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 3 feet from its fully inserted position within the reactor vessel.

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

4.9.1.1.3 Valve 3CHS-V305 shall be verified closed and locked 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.

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 of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal. A 1.6% $\Delta k/k$ SHUTDOWN MARGIN is required to provide protection against a boron dilution accident.

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.

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions of 1.6% Δk/k 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 a usable volume of 17,610 gallons of 6300 ppm borated water from the boric acid storage tanks or 1,166,000 gallons of 2700 ppm borated water from the refueling water storage tank (RWST). A minimum RWST volume of 1,166,000 gallons is specified to be consistent with ECCS requirement.

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 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 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.6% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either a usable volume of 1680 gallons of 6300 ppm borated water from the boric acid storage tanks or a usable volume of 85,840 gallons of 2700 ppm borated water from the RWST.

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 7.0 and 7.5 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 minimum RWST solution temperature for MODES 5 and 6 is based on analysis assumptions in addition to freeze protection considerations. The minimum/maximum RWST solution temperatures for MODES 1, 2, 3 and 4 are based on analysis assumptions.

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

MOVABLE CONTROL ASSEMBLIES (Continued)

rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ±12 steps at 24, 48, 120, and fully withdrawn position for the Control Banks and 18, 210, and fully withdrawn position for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). Trippability is defined in Attachment C to a letter dated December 21, 1984, from E. P. Rahe (Westinghouse) to C. O. Thomas (NRC). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately 4 hours for this verification.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to the design limit during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods; and

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the Core Operating Limits Report (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued)

At power levels below APL^{ND}, the limits on AFD are defined in the COLR consistent with the Relaxed Axial Offset Control (RAOC) operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible: (1) RAOC, the AFD limit of which are defined in the COLR, and (2) base load operation, which is defined as the maintenance of the AFD within COLR specifications band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(z)$ less than its limiting value. To allow operation at the maximum permissible power level, the base load operating procedure restricts the indicated AFD to relatively small target band (as specified in the COLR) and power swings ($APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For base load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the base load operation, a 24-hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the base load procedure. After the waiting period, extended base load operation is permissible.

The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are: (1) outside the allowed delta-I power operating space (for RAOC operation), or (2) outside the allowed delta-I target band (for base load operation). These alarms are active when power is greater than (1) 50% of RATED THERMAL POWER (for RAOC operation), or

AXIAL FLUX DIFFERENCE (Continued)

(2) APL^{ND} (for base load operation). Penalty deviation minutes for base load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ±12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 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 Conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

The $F^N_{\Delta H}$ as calculated in Specifications 3.2.3.1 and 3.2.3.2 are used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset any rod bow penalty and transition core penalty. The remaining margin is available for plant design flexibility.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height-dependent power factor appropriate to either RAOC or base load operation, W(z) or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$ is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by base load operation which result in less severe transient values. The W(z) and $W(z)_{BL}$ actions described above for normal operation are specified in the COLR per Specification 6.9.1.6.

When RCS flow rate and $F_{\Delta H}^N$ are measured, no additional allowances are necessary prior to comparison with the limits of the Limiting Condition for Operation. Measurement errors of 2.4% for four loop flow and 2.8% for three loop flow for RCS total flow rate and 4% for $F_{\Delta H}^N$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi will be added if venturis are inspected and cleaned at least once for 18 months. 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 the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

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

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation defined in Specifications 3.2.3.1 and 3.2.3.2.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the design limit throughout each analyzed transient. The indicated T_{avg} value of 591.1°F (four loop

DNB PARAMETERS (Continued)

operations) or 583.3°F (three loops operating) and the indicated pressurizer pressure value is 2218 psia (four loop or three loop operation). The calculated values of the DNB related parameters will be an average of the indicated values for the operable channels.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. Measurement uncertainties have been accounted for in determining the parameter limits.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate in MODES 1 and 2 with three or four reactor coolant loops in operation and maintain DNBR greater than the design limit during all normal operations and anticipated transients. With less than the required reactor coolant loops in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, three reactor coolant loops, and in Mode 4, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The requirement to maintain the isolated loop stop valves shut with power removed ensures that no reactivity addition to the core could occur due to the startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop. The 2600 ppm is sufficient to bound shutdown margin requirements and provide for boron concentration measurement uncertainty between the loop and the RWST. Draining and refilling the isolated loop within 4 hours prior to opening its stop valves ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

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. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2600 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The 2600 ppm provides for boron concentration measurement uncertainty between the spent fuel pool and the RWST. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.1.2 Boron Concentration in Spent Fuel Pool

The limitations of this specification ensure that in the event of a fuel assembly handling accident involving either a misplaced or dropped fuel assembly, the K_{eff} of the spent fuel storage racks will remain less than or equal to .95.

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.

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.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.4 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

A supplemental report containing dose assessments for the previous year shall be submitted annually within 90 days after January 1.

The report shall include that information delineated in the REMODCM.

Any changes to the REMODCM shall be submitted in the Semiannual Radioactive Effluent Release Report.

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator Region I, and one copy to the NRC Resident Inspector, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.6.a 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 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, target band, and APL^{ND} for Specifications 3/4.2.1.1 and 3/4.2.1.2,
5. Heat Flux Hot Channel Factor, K(z), W(z), APL^{ND}, and $W(z)_{BL}$ for Specifications 3/4.2.2.1 and 3/4.2.2.2.
6. Nuclear Enthalpy Rise Hot Channel Factor, Power Factor Multiplier for Specification 3/4.2.3.

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

6.9.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and 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, 3.2.3--Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1981 (W Proprietary).

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.

4. NUREG-800, 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), Revision 2, July 1981.

5. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary). (Methodology for Specifications 3.2.1--Axial Flux Difference [Relaxed Axial Offset Control] and 3.2.2--Heat Flux Hot Channel Factor [W(z) surveillance requirements for F_Q Methodology].)

6. WCAP-9561-P-A, ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS--SPECIAL REPORT: THIMBLE MODELING W ECCS EVALUATION MODEL," July 1986 (W Proprietary). (Methodology for Specification 3.2.2--Heat Flux Hot Channel Factor.)

7. WCAP-10266-P-A, Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987 (W Proprietary). (Methodology for Specification 3.2.2--Heat Flux Hot Channel Factor.)

8. WCAP-11946, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Millstone Nuclear Power Station Unit 3," September 1988 (W Proprietary).

6.9.1.6.c 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.

6.9.1.6.d 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.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RADIOACTIVE WASTE TREATMENT SYSTEMS

1.25 RADIOACTIVE WASTE TREATMENT SYSTEMS are those liquid, gaseous and solid waste systems which are required to maintain control over radioactive material in order to meet the Limiting Conditions for Operation (LCOs) set forth in these specifications.

RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

1.26 A RADIOLOGICAL EFFLUENT MONITORING MANUAL shall be a manual containing the site and environmental sampling and analysis programs for measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures to individuals from station operation. An OFFSITE DOSE CALCULATION MANUAL shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setponts. Requirements of the REMODCM are provided in Specification 6.13.

RATED THERMAL POWER

1.27 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.29 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

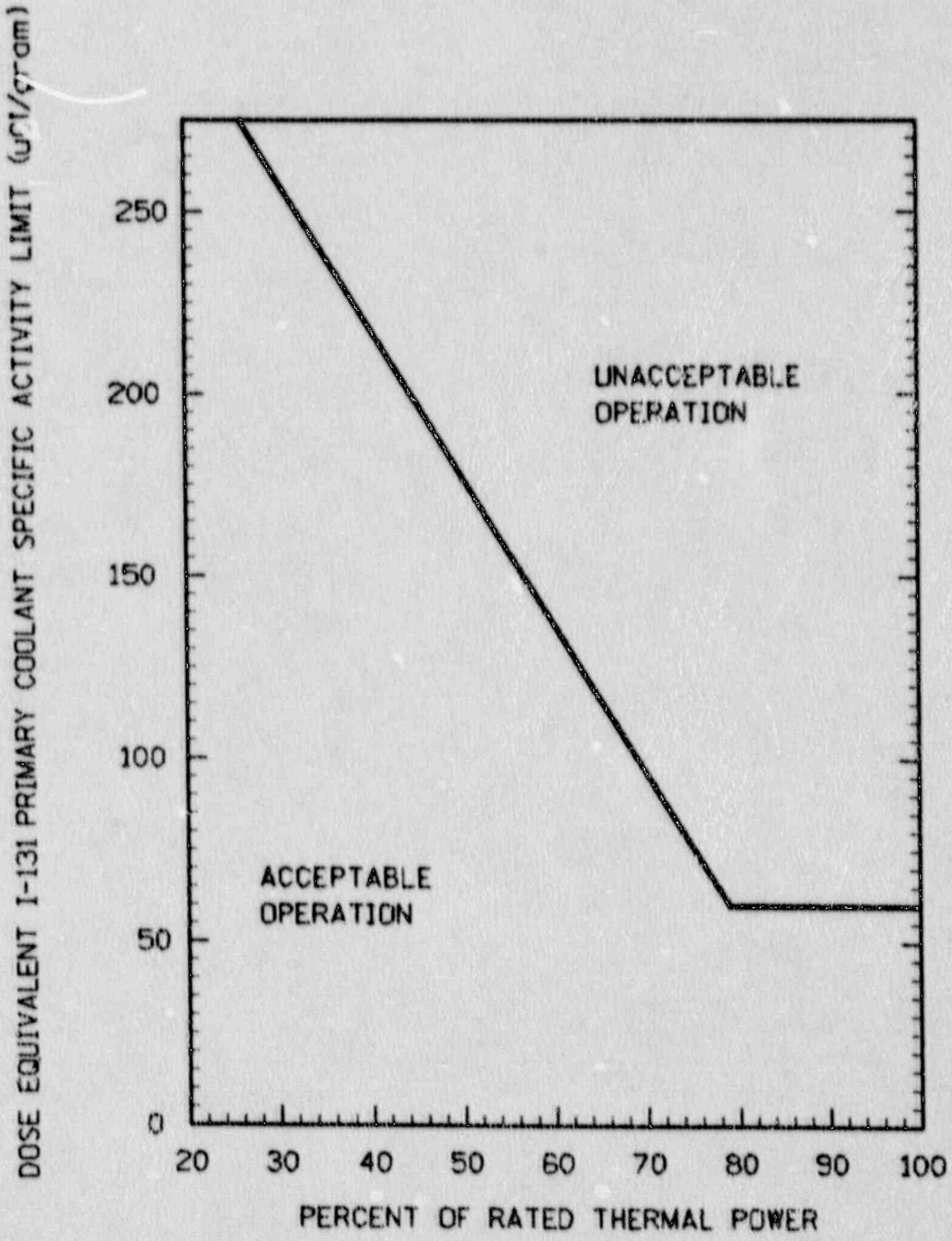

FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVIY >1 uCi/gram DOSE EQUIVALENT I-131

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	: PLATE METAL
COPPER CONTENT	: CONSERVATIVELY ASSUMED TO BE 0.10 WT %
PHOSPHORUS CONTENT	: 0.010 WT %
RT_{NDT} INITIAL	: 60°F
RT_{NDT} AFTER 10 EFPY	: 1/4T, 122°F 3/4T, 101°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

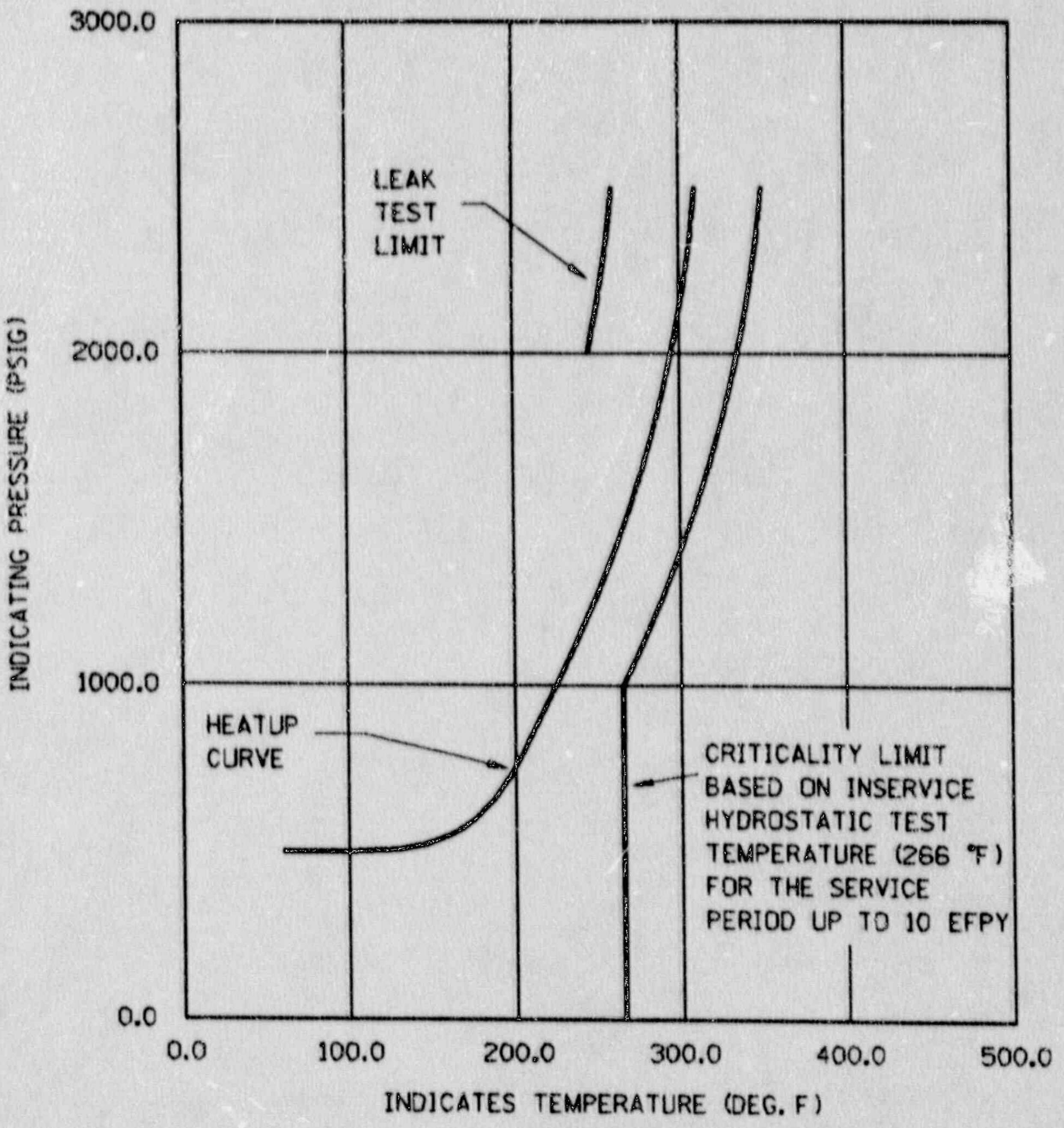

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 10 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	: PLATE METAL
COPPER CONTENT	: CONSERVATIVELY ASSUMED TO BE 0.10 WT %
PHOSPHORUS CONTENT	: 0.010 WT %
RT_{NDT} INITIAL	: 60°F
RT_{NDT} AFTER 10 EFPY	: 1/4T, 122°F 3/4T, 101°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

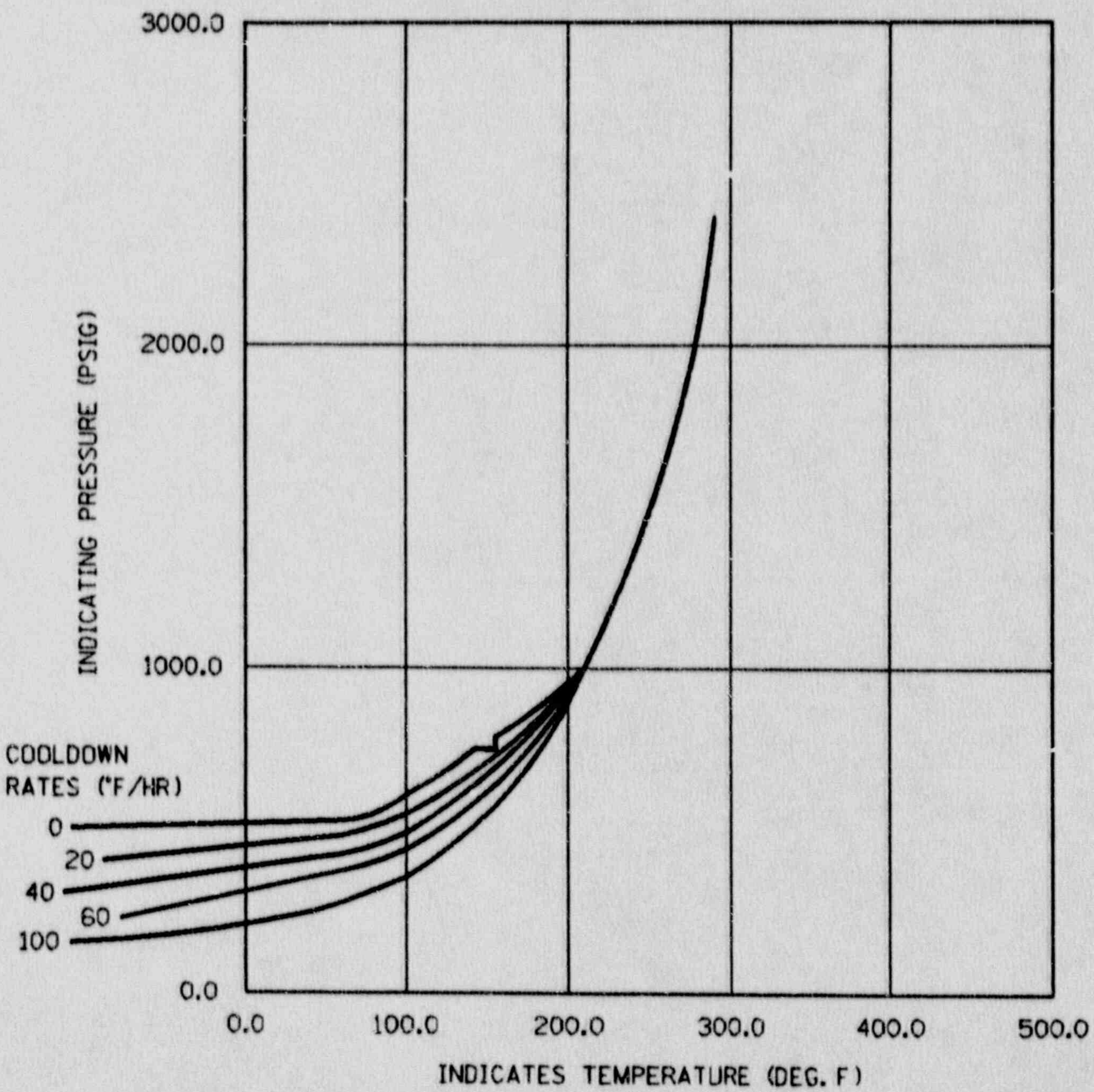

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 10 EFPY

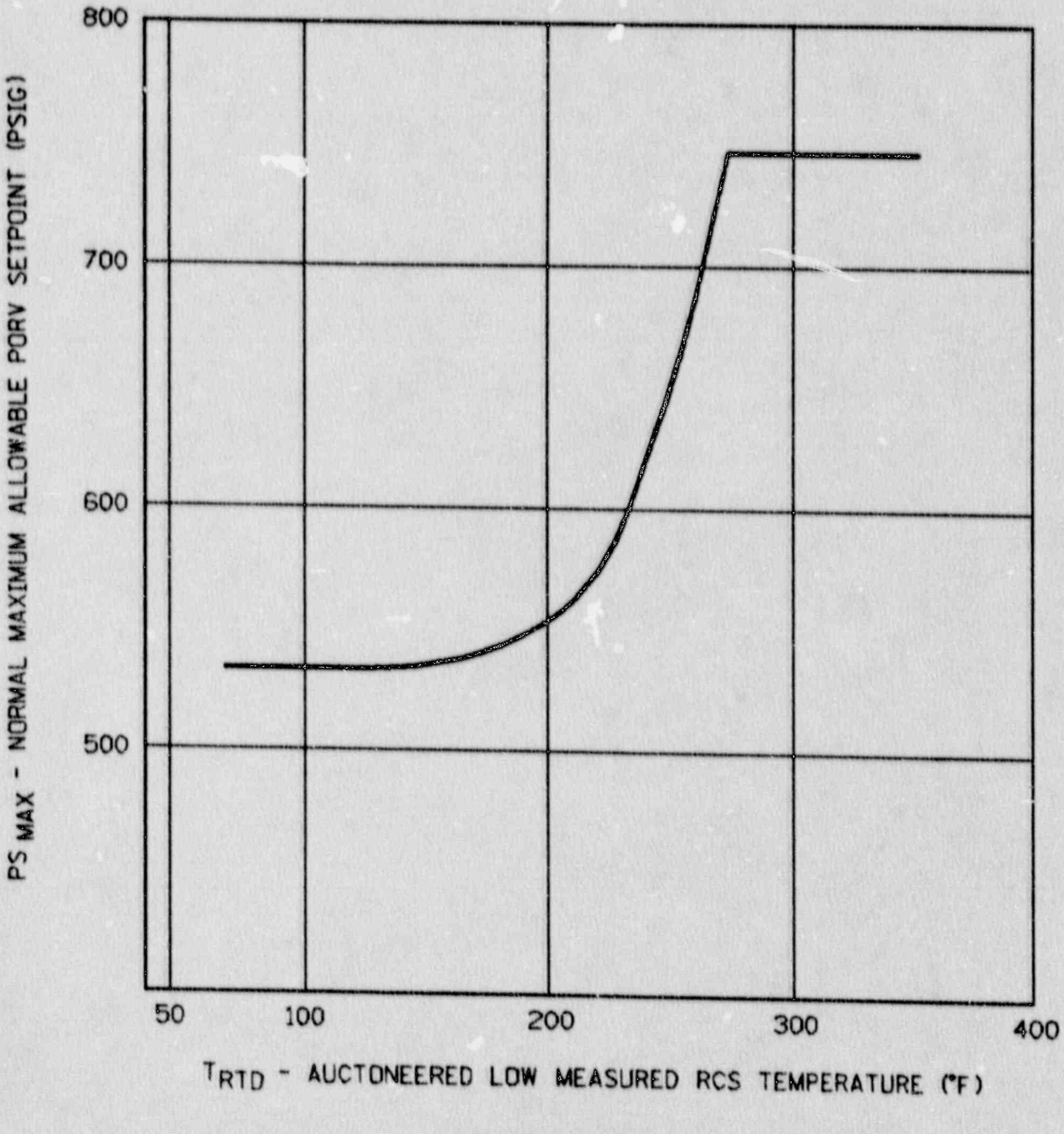

FIGURE 3.4-4a

NOMINAL MAXIMUM ALLOWABLE PORV
SETPOINT FOR THE COLD OVERPRESSURE SYSTEM
(FOUR LOOP OPERATION)

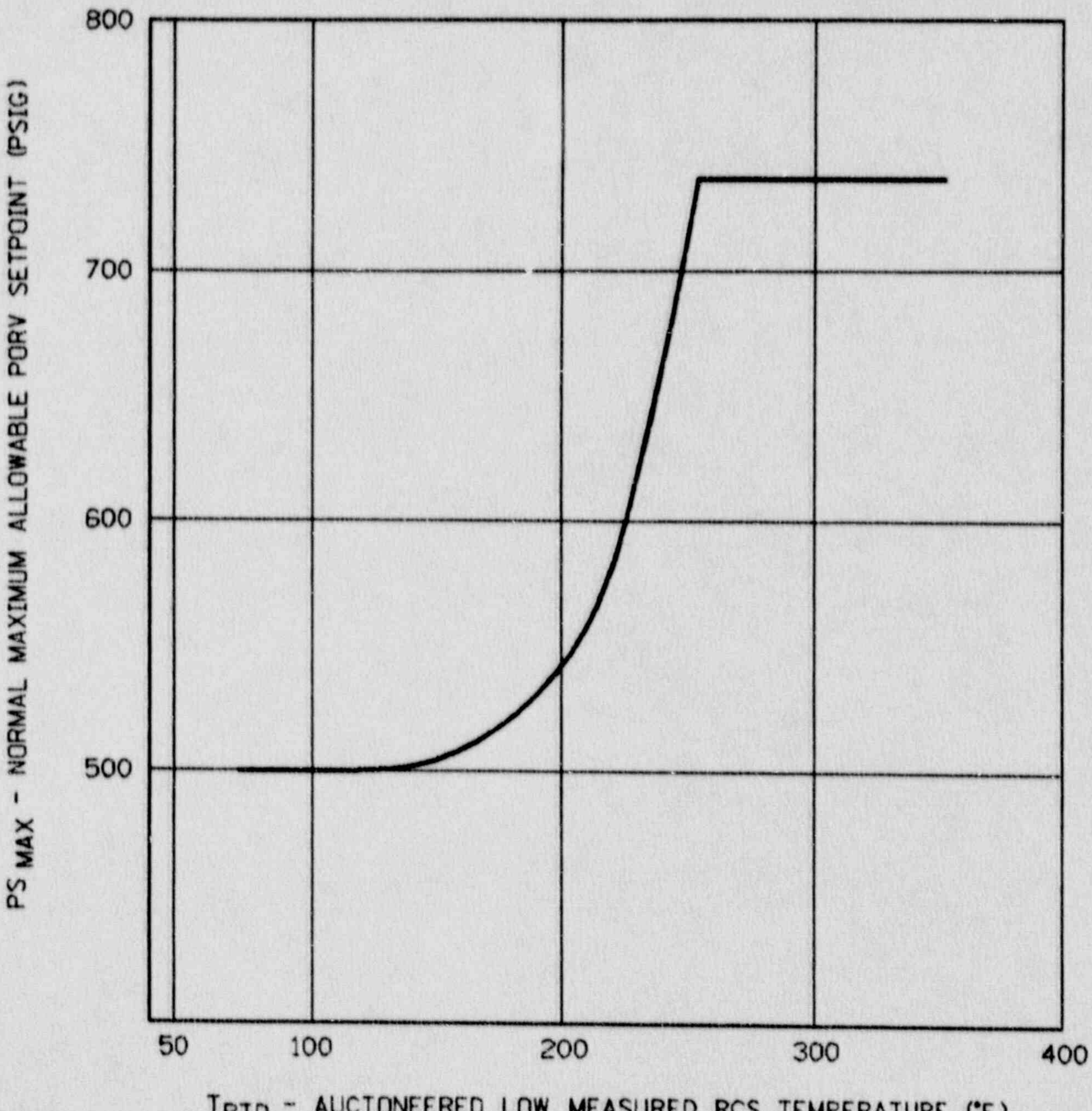

FIGURE 3.4-4b

NOMINAL MAXIMUM ALLOWABLE PORV
SETPOINT FOR THE COLD OVERPRESSURE SYSTEM
(THREE LOOP OPERATION)

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High 1	3.3	1.01	1.75	≤ 3.0 psig	≤ 3.8 psig
d. Pressurizer Pressure--Low					
1) Channels I and II	22.16	20.1	1.5	≥ 1877.3 psig	≥ 1868.5 psig
2) Channel III and IV	22.16	15.6	3.3	≥ 1877.3 psig	≥ 1863.3 psig
e. Steam Line Pressure--Low	17.7	15.6	2.2	≥ 658.6 psig*	≥ 648.3 psig*
2. Containment Spray (CDA)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3	3.3	1.01	1.75	≤ 8.0 psig	≤ 8.8 psig
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.

Attachment 3

Millstone Nuclear Power Station, Unit No. 3

Responses to Conditions Contained in
NRC SER on WCAP-10444

November 1990

Millstone Nuclear Power Station, Unit No. 3 Responses to Conditions Contained in NRC SER on WCAP-10444

The NRC Staff reviewed WCAP-10444, "Reference Core Report VANTAGE 5H Fuel Assembly," and concluded that the Westinghouse topical report was an acceptable reference to support plant-specific application of VANTAGE 5 provided certain conditions were addressed. The conditions are addressed for the Millstone Unit No. 3 application requesting the use of VANTAGE 5H fuel assemblies. These conditions are listed below with comments addressing their relevance to the VANTAGE 5H fuel assemblies' upgrade licensing submittal material in the report titled "The Plant Safety Evaluation for Millstone Unit No. 3, VANTAGE 5H Fuel Upgrades, August 1990," and the Technical Specification changes (Attachments 1 and 2).

Condition 1

The statistical convolution method described in WCAP-10125 for the evaluation of initial fuel rod to nozzle growth gap has not been approved. This method should not be used in VANTAGE 5.

Response

Worst-case fabrication tolerances and VANTAGE 5H fuel rod and assembly growth are used to determine the initial fuel rod to nozzle growth gaps in the evaluation of fuel rod performance summarized in Section 2 of the Plant Safety Evaluation (PSE) as per Section M of WCAP 10125-P-A.

Condition 2

For each plant application, it must be demonstrated that the LOCA/seismic loads considered in WCAP-9401 bound the plant in question; otherwise additional analysis will be required to demonstrate the fuel assembly structural integrity.

Response

Millstone Unit No. 3 is not completely bounded by the LOCA/seismic loads considered in WCAP-9401. Thus, in accordance with Condition 2 of the VANTAGE 5 NRC SER, additional plant-specific analyses have been performed to demonstrate fuel assembly structural integrity.

An evaluation of 17 x 17 VANTAGE 5H (with IFMs) fuel assembly structural integrity considering the lateral effects of seismic and LOCA accidents has been performed. In accordance with NRC requirements in Appendix A, Standard Review Plan (SRP) 4.2, the results show that the 17 x 17 VANTAGE 5H is structurally acceptable for an all VANTAGE 5H core and a transition core consisting of both VANTAGE 5H and LOPAR fuel. The grid load comparative study results

show that the LOPAR assembly loads bound the transition core composed of both VANTAGE 5H and LOPAR assemblies. The grids will not buckle due to combined impact forces of a seismic/LOCA event. The core coolable geometry is maintained. The stresses resulting from seismic- and LOCA-induced deflections are within acceptable limits. The reactor can be shut down under the combined faulted condition loads.

Condition 3

An irradiation demonstration program should be performed to provide early confirmation performance data for the VANTAGE 5 design.

Response

The VANTAGE 5H fuel assembly design is very similar to the VANTAGE 5 design for which a demonstration program was performed to determine early performance data on the VANTAGE 5 fuel assembly design features. The VANTAGE 5 Demonstration Program is described in Section 3.2.3 of Reference 2.

Condition 4

For those plants using the Improved Thermal Design Procedure (ITDP), the restrictions enumerated in Section 4.1 of this report must be addressed and information regarding measurement uncertainties must be provided.

Response

Westinghouse has addressed (via Reference 3) the restrictions enumerated in Section 4.1 of the VANTAGE 5 NRC SER. The RTDP instrument uncertainty methodology used for Millstone Unit No. 3 with VANTAGE 5H fuel is similar to that presented in WCAP-11656 (Reference 4) for ITDP. RTDP measurement uncertainties were provided to the NRC in Reference 5.

Condition 5

The WRB-2 correlation with a DNBR limit of 1.17 is acceptable for application to 17 x 17 VANTAGE 5 fuel. Additional data and analysis are required when applied to 14 x 14 or 15 x 15 fuel with an appropriate DNBR limit. The applicability range of WRB-2 is specified in Section 4.2.

Response

17 x 17 VANTAGE 5H fuel will be utilized at the Millstone Unit No. 3 plant. As described in Section 4 of Reference 1, the WRB-2 correlation with a DNBR limit of 1.17 is used for the VANTAGE 5H fuel with RTDP methodology.

Condition 6

For 14 x 14 and 15 x 15 VANTAGE 5 fuel designs, separate analyses will be required to determine a transitional mixed-core penalty. The mixed core penalty and plant-specific safety margin to compensate for the penalty should be addressed in the plant Technical Specification bases.

Response

17 x 17 VANTAGE 5H fuel will be utilized at Millstone Unit No. 3. As stated in Section 4.6 of the PSE, "Transition cores are analyzed as if they were full cores of one assembly type (full STD or full VANTAGE 5H), applying the applicable transition core penalty. For VANTAGE 5 fuel, penalties are a function of the number of VANTAGE 5 fuel assemblies in the core, as per Reference 6, which has been approved by the NRC. The same methodology is used to calculate VANTAGE 5H transition core penalties. The DNBR penalty is less than 12.5%."

The transition core penalty is covered by the margin maintained between the design and safety limit DNBR. This margin is addressed in the proposed changes to Section 2.1.1 of the Millstone Unit No. 3 Technical Specifications, Reactor Core Limit Bases. Maintenance of adequate DNBR margin to cover DNBR penalties is confirmed on a cycle-specific basis during the reload safety evaluation process.

Condition 7

Plant-specific analysis should be performed to show that the DNBR limit will not be violated with the higher valve of $F_{\Delta H}$.

Response

The core DNB methodology as applied to Millstone Unit 3 with VANTAGE 5H fuel is presented in Section 4 of the PSE. The PSE contains Millstone Unit No. 3-specific analyses which support the use of an $F_{\Delta H}$ of 1.70 (with appropriate treatment of uncertainties) during the transition period with a full core of VANTAGE 5H fuel. All safety criteria are met with an $F_{\Delta H}$ of 1.70 at 100 percent rated power (3411 MWt reactor power).

Condition 8

The plant-specific safety analysis for the steam system piping failure event should be performed with the assumption of loss of power if that is the most conservative case.

Response

A specific safety evaluation for Major Secondary Steam System Pipe Rupture was performed in support of the Millstone Unit No. 3 transition to VANTAGE 5H fuel. This evaluation is described in Section 5.1.3.5 of the PSE.

Condition 9

With regard to the RCS pump shaft seizure accident, the fuel failure criterion should be the 95/95 DNBR limit. The mechanistic method mentioned in WCAP-10444 is not acceptable.

Response

The mechanistic method was not used with regard to the RCS pump shaft seizure (locked rotor) accident addressed in Section 5.1.5.3 of the PSE. Any rods which violated the 95/95 DNBR limit are assumed to fail.

Condition 10

If a positive MTC is intended for VANTAGE 5, the same positive MTC consistent with the plant Technical Specifications should be used in the plant-specific safety analysis.

Response

In conjunction with the transition to VANTAGE 5H fuel, a positive MTC is used for Millstone Unit No. 3 as described in Section 1 of the PSE. The supporting Millstone Unit No. 3 safety analyses have utilized MTC assumptions which are consistent with or conservative with respect to the current Technical Specifications. Specific discussion of MTC assumptions within the VANTAGE 5H safety analysis is presented in Section 5.1.1.2 of the PSE. All non-LOCA accidents reanalyzed demonstrate that the appropriate safety criteria are met.

Condition 11

The LOCA analysis performed for the reference plant with higher F_Q of 2.55 has shown that the PCT limit of 2200°F is violated during a transitional mixed core. Plant-specific LOCA analysis must be done to show that with the appropriate value of F_Q, the 2200°F criterion can be met during use of a transitional mixed core.

Response

In accordance with Condition 11 of the VANTAGE 5 NRC SER, Millstone Unit No. 3-specific LOCA analysis for cores fueled with VANTAGE 5H and Standard assemblies were performed with consideration of transitional core effects. The large- and small-break LOCA analyses are presented in Section 5.2.1 and 5.2.2 of the PSE. As described therein, the ECCS acceptance criterion of 2200°F is met for Millstone Unit No. 3 with a LOCA F_Q of 2.60 for four-loop operation and 3.00 for three-loop operation. The worst-case peak clad temperature is 2133°F for a four-loop LOCA and 1874°F for the three-loop case. A conservative transition core penalty of 50°F has been applied to the PCT of the VANTAGE 5H assembly.

Condition 12

Our SER on Westinghouse's extended burnup topical report WCAP-10125 is not yet complete; the approval of the VANTAGE 5 design for operation to extended burnup levels is contingent on NRC approval of WCAP-10125. However, VANTAGE 5 fuel may be used to those burnups to which Westinghouse fuel is presently operating. Our review of the Westinghouse extended burnup topical report has not identified any safety issues with operation to the burnup value given in the extended burnup report.

Response

WCAP-10125 has been approved (see Reference 1). The extended burnup methodology contained in this topical has been applied and is considered in Section 2 of the PSE.

Condition 13

Recently, a vibration problem has been reported in a French reactor having 14-foot fuel assemblies: vibration below the fuel assemblies in the lower position of the reactor vessel is damaging the movable incore instrumentation probe thimble. The staff is currently evaluating the implications of this problem to other cores having 14-foot-long bundle assemblies. Any limitations to the 14-foot core design resulting from the staff evaluation must be addressed in plant-specific evaluations.

Response

Millstone Unit No. 3 has 12-foot-long fuel assembly bundles and therefore the above condition is not applicable. However, NNECO has established a program to monitor incore thimble tube degradation through use of eddy-current testing. NNECO has implemented all NRC required testing (IE Bulletin 88-09) of the thimble tubes.

References

1. Davidson, S. L., (Ed.) et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A (Proprietary), December 1985.

2. Skaritka, J., "Operational Experience with Westinghouse Core," WCAP-8183, Revision 17, August 1989.

3. Westinghouse letter, E. P. Rahe, Jr., to C. O. Thomas (NRC), "Response to Request Number 1 for Additional Information on WCAP-10444 entitled VANTAGE 5 Fuel Assembly" (Proprietary), NS-NRC-85-3014, dated March 1, 1985.

4. "Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology," WCAP-11656, December 1987.

5. Letter from NEU to NRC, Transmittal of WCAP-12621, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Northeast Utilities, Millstone Unit 3 Nuclear Power Station," August 1990.

6. P. Schueren and Mr. K. R. McAtee, "Extension of Methodology for Calculating Transition Core DNBR Penalties," WCAP-11837-P-A, January 1990.