ATTACHMENT 1

SAFETY LIMITS AND POWER DISTRIBUTION

TS MARK-UPS

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1a for four loop operation.
(Unit 1) and 2.1-1b (Unit 2)

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

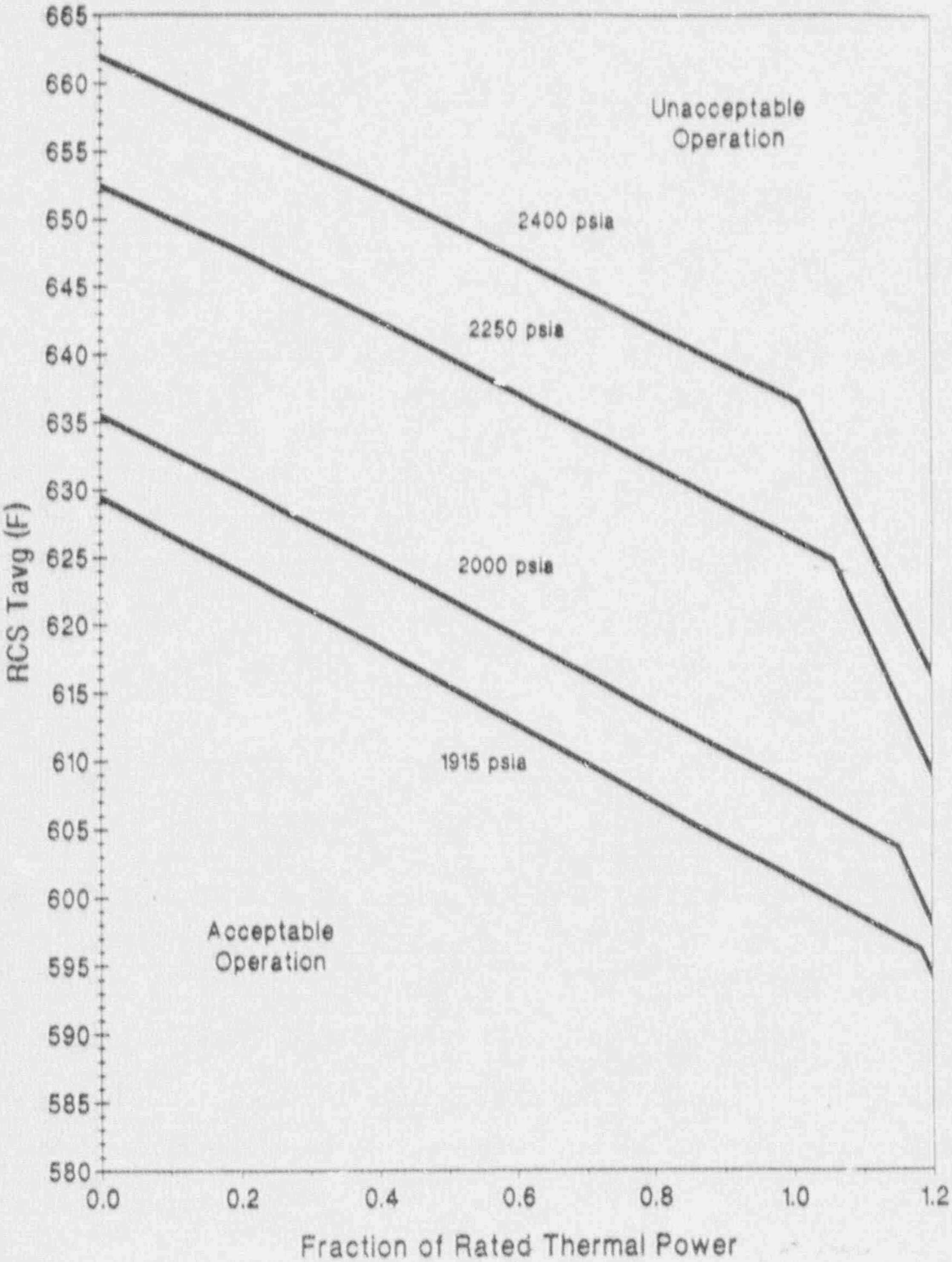
MODES 1 and 2:

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

MODES 3, 4, and 5:

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

Figure 2.1-1a Reactor Core Safety Limits - Four Loops in Operation (UNIT 1)



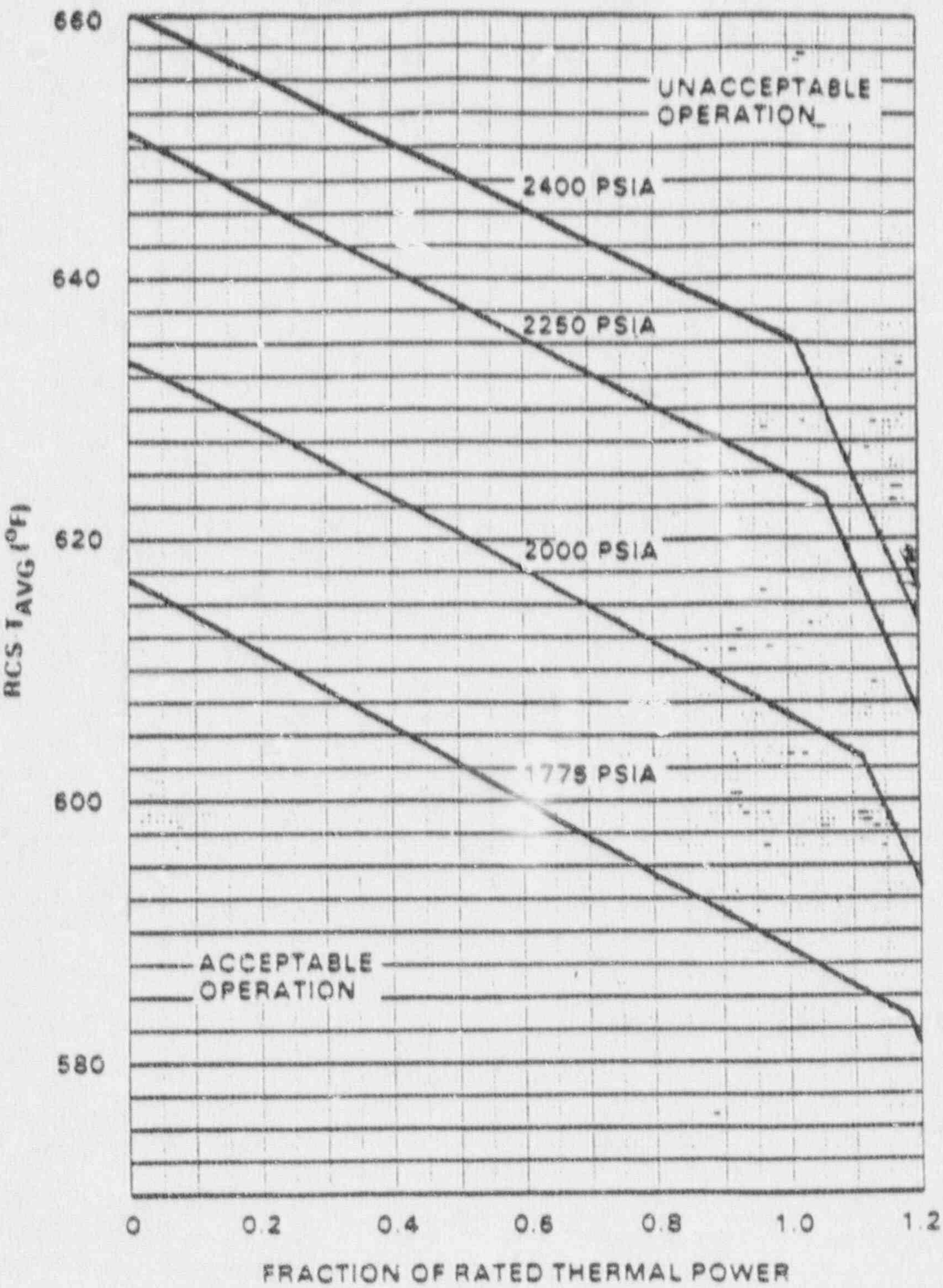


FIGURE 2.1-1 b

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION UNIT 2

TABLE 2.2.-1
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	~~4.56~~ 5.92	0	≤109% of RTP*	≤~~111.1%~~ 110.9% of RTP*
b. Low Setpoint	8.3	~~4.56~~ 5.92	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	8.4	0	≤25% of RTP*	≤31% of RTP*
6. Source Range, Neutron Flux	17.0	10	0	≤10^5 cps	≤1.4 x 10^5 cps
7. Overtemperature ΔT	~~7.2 (8.9#)~~ 6.98	~~4.41 (5.41#)~~ 3.0	~~2.03 (2.65#)~~ 2.12	See Note 1	See Note 2
8. Overpower ΔT	~~4.3 (4.9#)~~	~~1.3 (1.24#)~~	~~1.2 (1.7#)~~	See Note 3	See Note 4
9. Pressurizer Pressure-Low	4.0	2.21	1.5	≥1945 psig	≥1938 psig***
10. Pressurizer Pressure-High	7.5	~~4.96~~ 0.71	0.5	≤2385 psig	≤2399 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	~~2.5~~ 2.92	~~[illegible] (1.41#)~~ 1.48	0.6	≥90% of loop minimum measured flow**	~~≥89.2% (88.8%#)~~ 88.9 of loop minimum measured flow**

*RTP = RATED THERMAL POWER

**Loop minimum measured flow = 96,900 gpm (Unit 2), 96,250 gpm (Unit 1)

***Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to these values.

~~#A[illegible]cable upon deletion of RTD Bypass System.~~

No Changes to this page

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low					
a. Unit 1	17	14.2	1.5	≥17% of span from 0% to 30% RTP* increasing linearly to ≥ 40.0% of span from 30% to 100% RTP*	≥15.3% of span from 0% to 30% RTP* increasing linearly to ≥38.3% of span from 30% to 100% RTP*
b. Unit 2	11.8	1.7	2.0	≥36.8% of narrow range span	≥35.1% of narrow range span
14. Undervoltage - Reactor Coolant Pumps	8.57	0	1.0	≥77% of bus voltage (5082 volts) with a 0.7s response time	≥76% (5016 volts)
15. Underfrequency - Reactor Coolant Pumps	4.0	0	1.0	≥56.4 Hz with a 0.2s response time	≥55.9 Hz
16. Turbine Trip					
a. Stop Valve EH Pressure Low	N.A.	N.A.	N.A.	≥550 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP RATED [illegible]

No Changes to this page

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤10% of RTP*	≤12.2% of RTP*
2) P-13 input	N.A.	N.A.	N.A.	<10% RTP* Turbine Impulse Pressure Equivalent	<12.2% RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤48% of RTP*	≤50.2% of RTP*
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	≤69% of RTP*	≤70% of RTP*
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A	≥10% of RTP*	≥7.8% of RTP*
f. Power Range Neutron Flux, Not P-10	N.A.	N.A.	N.A.	≤10% of RTP*	≤12.2% of RTP*
g. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	<10% RTP* Turbine Impulse Pressure Equivalent	<12.2% RTP* Turbine Impulse Pressure Equivalent
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.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT = Measured ΔT by ~~RTD Manifold Instrumentation~~ LOOP NARROW RANGE RTDs;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = ~~[illegible]~~ 12 s, τ_2 = 3 s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = ~~1.41~~ 1.38;

K_2 = 0.02401/°F;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = ~~[illegible]~~ 22 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, $\tau_6 = 0$.

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

NOTE 1: (Continued)

T'	$\leq$	590.8°F (Nominal T_{avg} allowed by Safety Analysis);
K_3	=	0.001189;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

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

(i) For $q_t - q_b$ between -22.5% and -6.5%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ is more negative than -22.5%, the ΔI Trip Setpoint shall be automatically reduced by 3.151% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ is more positive than -6.5%, the ΔI Trip Setpoint shall be automatically reduced by 1.641% 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 ~~2.4%~~ (3.0%).

No Changes to this page

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

NOTE 3: OVERPOWER ΔT

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

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 = 1.0704,

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 controller for T_{avg} dynamic compensation,

τ_7 = Time constant utilized in the rate-lag controller 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 = 0.001707/°F for $T > 590.8$°F and $K_6 = 0$ for $T \leq 590.8$°F,

T = As defined in Note 1,

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 590.8°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 ~~2.6%~~ 2.8%.

~~#Applicable upon deletion of RTD Bypass System~~

Unit 1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the acceptable limits specified in the CORE OPERATING LIMITS REPORT (COLR)

Delete

a. the allowed operational space as specified in the CORE OPERATING LIMITS REPORT (COLR) for RAOC operation, or

b. within the target band specified in the COLR about the target flux difference during baseload operation.

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

ACTION:

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

1. Either restore the indicated AFD to within the COLR 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.

Delete

b. For Base Load operation above APL^{ND}** with the indicated AXIAL FLUX DIFFERENCE outside of the applicable target band about the target flux difference:

1. Either restore the indicated AFD to within the COLR specified target band limits 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.

b ~~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.

Delete

*See Special Test Exceptions Specification 3.10.2.

**APL^{ND} is the minimum allowable (nuclear design) power level for base load operation and is specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

Unit 1

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.2.1.1 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:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

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.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least two OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, the target axial 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.

4.2.1.4 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 3/4.2.2 or by linear interpolation between the most recently measured values and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

Delete

Unit 1

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$ $F_Q(x,y,z)$

LIMITING CONDITION FOR OPERATION

$\bar{F}_Q(x,y,z)$ imposing

3.2.2 $F_Q(Z)$ shall be limited by imposing the following relationships:

$\bar{F}_Q^{MA}(x,y,z)$

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

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

Where: F_Q^{RTP} = the F_Q Limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

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

$F_Q(x,y,z)$ limit

K(Z) = the normalized $F_Q(Z)$ ~~for a given core height~~ specified in the COLR for the appropriate fuel types.

APPLICABILITY: MODE 1. (Unit 1)

ACTION:

$F_Q(x,y,z)$

Replace with Attachment 1

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 (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit, and

d ~~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.

$F_Q(x,y,z)$

$\bar{F}_Q^{MA}(x,y,z)$ = the measured heat flux hot channel factor $F_Q^M(x,y,z)$, with adjustments as specified in 4.2.2.3,

Attachment 1:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours, and

b. Control the AFD to within new AFD limits which are determined by reducing the allowable power at each point along the AFD limit lines of Specification 3.2.1 at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and reset the AFD alarm setpoints to the modified limits within 8 hours, and

c. POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit, and

Unit 1

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

$F_Q^M(x,y,z)^{(1)}$ whether $F_Q(x,y,z)$

4.2.2.2 ~~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.

Delete

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 are satisfied.

c. Satisfying the following relationship:

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

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

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 per Specification 6.9.1.9.

Replace with Attachment 2

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

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

*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 a power distribution map obtained.

Attachment 2:

b. Measuring $F_Q^M(X,Y,Z)$ at the earliest of:

1. At least once per 31 Effective Full Power Days, or

2. Upon reaching equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last determined[2], or

3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

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[1]No additional uncertainties are required in the following equations for $F_Q^M(X,Y,Z)$, because the limits include uncertainties.

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

Unit 1

Replace with Attachment 3

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \quad \frac{F_Q^M(z)}{K(z)}$$

has increased 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 4.2.2.2c., or

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

$$\underset{\text{over } z}{\text{maximum}} \quad \frac{F_Q^M(z)}{K(z)} \quad \text{is not increasing.}$$

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

1) Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

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

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

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 AFD limits of Specification 3.2.1 by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

Delete

c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

Attachment 3:

c. Performing the following calculations:

1. For each location, calculate the % margin to the maximum allowable design as follows:

$$\% \text{ Operational Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{OP}} \right) \times 100\%$$

$$\% \text{ RPS Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{RPS}} \right) \times 100\%$$

where $[F_Q^L(X,Y,Z]^{OP}$ and $[F_Q^L(X,Y,Z]^{RPS}$ are the Operational and RPS design peaking limits defined in the COLR.

2. Find the minimum Operational Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then either of the following actions shall be taken:

(a) Within 15 minutes:

(1) Control the AFD to within new AFD limits that are determined by:

$$(\text{AFD Limit})^{\text{reduced}}_{\text{negative}} = (\text{AFD Limit})^{\text{COLR}^{(3)}}_{\text{negative}} + [\text{NSLOPE}_1^{(3)} \times \text{Margin}^{\text{min}}_{\text{OP}}]_{\text{absolute value}}$$

$$(\text{AFD Limit})^{\text{reduced}}_{\text{positive}} = (\text{AFD Limit})^{\text{COLR}^{(3)}}_{\text{positive}} - [\text{PSLOPE}_1^{(3)} \times \text{Margin}^{\text{min}}_{\text{OP}}]_{\text{absolute value}}$$

where $\text{Margin}^{\text{min}}_{\text{OP}}$ is the minimum margin from 4.2.2.2.c.1, and

(2) Within 8 hours, reset the AFD alarm setpoints to the modified limits of 4.2.2.2.c.2.a, or

(b) Comply with the ACTION requirements of Specification 3.2.2.

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[3] Defined and specified in the COLR per Specification 6.9.1.9.

Attachment 3 (con't):

3. Find the minimum RPS Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then the following action shall be taken:

Within 72 hours, reduce the K_1 value for OTΔT by:

$$K_1^{adjusted} = K_1^{(4)} - [KSLOPE^{(3)} \times Margin^{min}_{RPS}]_{absolute\ value}$$

where $MARGIN^{min}_{RPS}$ is the minimum margin from 4.2.2.2.c.1.

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(3) Defined and specified in the COLR per Specification 6.9.1.9.

(4) K_1 value from Table 2.2-1.

Attachment 5 (cont'd)

d. Extrapolating the two most recent measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$[F_Q^M(X,Y,Z)]$ (extrapolated) $\geq [F_Q^L(X,Y,Z)]^{OP}$ (extrapolated), or

$[F_Q^M(X,Y,Z)]$ (extrapolated) $\geq [F_Q^L(X,Y,Z)]^{RPS}$ (extrapolated),

either of the following actions shall be taken:

1. $F_Q^M(X,Y,Z)$ shall be increased by 2 percent over that specified in 4.2.2.2.a, and the calculations of 4.2.2.2.c repeated, or

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.2.2.c.1 shall be performed no later than the time at which the margin in 4.2.2.2.c.1 is extrapolated to be equal to zero.

Unit 1

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

Replace with Attachment 4

g. The limits specified in Specifications 4.2.2.2c., 4.2.2.2e., and 4.2.2.2f., 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.

Delete

4.2.2.3 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 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within the target band about the target flux difference of Specification 3.2.1) 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 FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \underset{\text{over } Z}{\text{minimum}} \left[\frac{F_Q^{RTP} \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\%$$

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. $W(z)_{BL}$ 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 CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3a shall be satisfied before re-entering Base Load operation.

4.2.2.4 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 are satisfied.

---

*APL^{ND} is the minimum allowable (nuclear design) power level for Base Load operation in Specification 3.2.1.

.ttachment 4 :

e. The limits in Specifications 4.2.2.2.c and 4.2.2.2.d are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

Unit 1

Delete

c. Satisfying the following relationship:

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

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. 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. $W(Z)_{BL}$ 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 CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

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 surveillance 4.2.2.3 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 measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)}\right]$$

has increased since the previous determination $F_Q^M(Z)$ either of the following actions shall be taken:

1. $F_Q^M(Z)$ shall be increased by 2 percent over that specified in 4.2.2.4c, or

2. $F_Q^M(Z)$ shall be measured at least once per 7 EFPD until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)}\right] \text{ is not increasing.}$$

f. With the relationship specified in 4.2.2.4c above not being satisfied, either of the following actions shall be taken:

1. Place the core in an equilibrium condition where the limit in 4.2.2.2c is satisfied, and remeasure $F_Q^M(Z)$, or

Unit 1

SURVEILLANCE REQUIREMENTS (Continued)

Delete

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated with the following expression:

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

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

1. Lower core region 0 to 15 percent, inclusive.

2. Upper core region 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 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.

Replace with Attachment 5

Attachment 5:

4.2.2.3 When a full core power distribution map is taken for reasons other than meeting the requirements of Specification 4.2.2.2, an overall $F_Q^M(X,Y,Z)$ shall be determined, then increased by 3% to account for manufacturing tolerances, further increased by 5% to account for measurement uncertainty, and further increased by the radial-local peaking factor to obtain a maximum local peak. This value shall be compared to the limit in Specification 3.2.2.

Unit 1

SURVEILLANCE REQUIREMENTS

4.2.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 F_{xy} 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_{xy} 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,

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1+0.2(1-P)],$$

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

Delete

Unit 1

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.9;

f. The F_{xy} limits of Specification 4.2.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, 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.

Delete

Unit 1

3/4.2.3 ~~REACTOR COOLANT SYSTEM FLOW RATE AND~~ NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

Replace with Attachment 1

3.2.3 The combination of indicated Reactor Coolant System total flow rate and R shall be maintained within the region of permissible operation specified in the CORE OPERATING LIMITS REPORT (COLR) for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{F^{RTP}_{\Delta H}[1.0 + MF_{\Delta H}(1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since the figure specified in the COLR includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 2.1% for flow and 4% for incore measurement of $F^N_{\Delta H}$,

d. $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

e. $MF_{\Delta H}$ = The power factor multiplier specified in the COLR.

APPLICABILITY: MODE 1. (Unit 1)

ACTION:

Replace with Attachment 2

a. With the combination of Reactor Coolant System total flow rate and R within the region of restricted operation within 6 hours reduce the Power Range Neutron Flux-High Trip Setpoint to below the nominal setpoint by the same amount (% RTP) as the power reduction required by the figure specified in the COLR.

b. With the combination of Reactor Coolant System total flow rate and R within the region of prohibited operation specified in the COLR:

1. Within 2 hours either:

    a) Restore the combination of Reactor Coolant System total flow rate and R to within the region of permissible operation, or

    b) Restore the combination of Reactor Coolant System total flow rate and R to within the region of restricted operation and comply with action a. above, or

Attachment 1:

3.2.3 $F_{\Delta H}(X,Y)$ shall be limited by imposing the following relationship:

$F\Delta HR^{M}(X,Y) \leq F\Delta HR^{L}(X,Y)$

Where: $F\Delta HR^{M}(X,Y)$ = the maximum measured radial peak ratio as defined in the CORE OPERATING LIMITS REPORT (COLR).

$F\Delta HR^{L}(X,Y)$ = the maximum allowable radial peak ratio as defined in the COLR.

Attachment 2:

ACTION:

With $F_{\Delta H}(X,Y)$ exceeding its limit:

a. Within 2 hours, reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH%[1] for each 1% that $F\Delta HR^M(X,Y)$ exceeds the limit, and

b. Within 6 hours either:

1. Restore $F\Delta HR^M(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

2. Reduce the Power Range Neutron Flux-High Trip Setpoint in Table 2.2-1 at least RRH% for each 1% that $F\Delta HR^M(X,Y)$ exceeds that limit, and

c. Within 72 hours of initially being outside the limit of Specification 3.2.3, either:

1. Restore $F\Delta HR^M(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

2. Perform the following actions:

(a) Reduce the OTΔT K_1 term in Table 2.2-1 by at least TRH[2] for each 1% that $F\Delta HR^M(X,Y)$ exceeds the limit, and

(b) Verify through incore mapping that $F\Delta HR^M(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

---

[1] RRH is the amount of THERMAL POWER reduction required to compensate for each 1% that $F\Delta HR^M(X,Y)$ exceeds $F\Delta HR^L(X,Y)$, provided in the COLR per Specification 6.9.1.9.

[2] TRH is the amount of OTΔT K_1 setpoint reduction required to compensate for each 1% that $F\Delta HR^M(X,Y)$ exceeds the limit of Specification 3.2.3, provided in the COLR per Specification 6.9.1.9.

Unit 1

3/4.2.3 ~~REACTOR COOLANT SYSTEM FLOW RATE AND~~ NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(x,y)$

LIMITING CONDITION FOR OPERATION

Delete (Incorporated in Specification 3.2.5)

ACTION (Continued)

c) 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.

2. Within 24 hours of initially being within the region of prohibited operation specified in the COLR, verify through incore flux mapping and Reactor Coolant System total flow rate comparison that the combination of R and Reactor Coolant System total flow rate are restored to within the regions of restricted or permissible operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

Delete

Unit 1

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

a. and/or c.2.

d ~~f~~. 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 b.1.c) and/or b.2., above; subsequent POWER OPERATION may proceed provided that ~~the combination of R and indicated Reactor Coolant System total flow rate are~~ $F\Delta H R^{M}(X,Y)$ demonstrated, through incore flux mapping ~~and Reactor Coolant System total flow rate comparison~~, to be within the limit ~~regions of restricted or permissible operation~~ specified in the COLR prior to exceeding the following THERMAL POWER levels:

1 ~~a~~) ~~A nominal~~ 50% of RATED THERMAL POWER,

2 ~~b~~) ~~A nominal~~ 75% of RATED THERMAL POWER, and

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

Insert Attachment 4

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

Replace with Attachment 3

4.2.3.2 The combination of indicated Reactor Coolant System total flow rate determined by process computer readings or digital voltmeter measurement and R shall be determined to be within the regions of restricted or permissible operation specified in the COLR:

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.

4.2.3.3 The indicated Reactor Coolant System total flow rate shall be verified to be within the regions of restricted or permissible operation specified in the COLR at least once per 12 hours when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The Reactor Coolant System 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.

4.2.3.5 The Reactor Coolant System total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

Delete (Incorporated in Specification 4.2.5)

Insert Attachment 5

Attachment 3:

b. Measuring $F\Delta HR^{M}(X,Y)$ according to the following schedule:

1. Prior to operation above 75% of RATED THERMAL POWER at the beginning of each fuel cycle, and the earlier of:

2. At least once per 31 Effective Full Power Days, or

3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

Attachment 4:

4.2.3.2 $F\Delta HR^{M}(X,Y)$ shall be evaluated to determine whether $F_{\Delta H}(X,Y)$ 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.

Attachment 5:

c. Performing the following calculations:

1. For each location, calculate the % margin to the maximum allowable design as follows:

$$\%F_{\Delta H}\text{ Margin} = (1 - \frac{F\Delta HR^{M}(X,Y)}{F\Delta HR^{L}(X,Y)}) \times 100\%$$

No additional uncertainties are required for $F\Delta HR^{M}(X,Y)$, because $F\Delta HR^{L}(X,Y)$ includes uncertainties.

2. Find the minimum margin of all locations examined in 4.2.3.2.c.1 above. If any margin is less than zero, comply with the ACTION requirements of Specification 3.2.3.

d. Extrapolating the two most recent measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$F\Delta HR^{M}$ (extrapolated) $\geq$ $F\Delta HR^{L}$ (extrapolated)

either of the following actions shall be taken:

1. $F\Delta HR^{M}(X,Y)$ shall be increased by 2 percent over that specified in 4.2.3.2.a, and the calculations of 4.2.3.2.c repeated, or

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.3.2.c shall be performed no later than the time at which the margin in 4.2.3.2.c is extrapolated to be equal to zero.

Unit 1

POWER DISTRIBUTION LIMITS

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 above 50% of RATED THERMAL POWER. (Unit 1)

APPLICABILITY: MODE 1*,**

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~~ 1.02 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.

Insert Attachment 1

for Specification 3.2.4

Attachment 1:

**Not applicable until calibration of the excore detectors is completed subsequent to refueling.

Unit 1

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

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 1.02 ~~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.

Unit 1

LIMITING CONDITION FOR OPERATION

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.

c. The provisions of Specification 4.0.4 are not applicable.

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.

Unit 1

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}, ~~and~~

b. Pressurizer Pressure,

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1.(unit 1)

ACTION:

a. With ~~any~~ either of the ~~above~~ parameters identified in 3.2.5 a. and b. above 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.

Add Insert ①, attached

SURVEILLANCE REQUIREMENTS

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

Add Insert ②, attached

Unit 1

Insert ①

~~a.~~ b. With the combination of Reactor Coolant System total flow rate and R THERMAL POWER within the region of restricted operation specified on Figure 3.2-1, within 6 hours reduce the Power Range Neutron Flux-High Trip Setpoint to below the nominal setpoint by the same amount (% RTP) as the power reduction required by ~~the figure specified in the COLR.~~ Figure 3.2-1.

~~b.~~ c. With the combination of Reactor Coolant System total flow rate and ~~R~~ THERMAL POWER within the region of prohibited operation specified ~~in the COLR~~ on Figure 3.2-1:

1. Within 2 hours either:

   a) Restore the combination of Reactor Coolant System total flow rate and ~~R~~ THERMAL POWER to within the region of permissible operation, or

   b) Restore the combination of Reactor Coolant System total flow rate and ~~R~~ THERMAL POWER to within the region of restricted operation and comply with action a. above, or

   c) 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.

2. Within 24 hours of initially being within the region of prohibited operation specified ~~in the COLR~~ on Figure 3.2-1, verify ~~through incore flux mapping and Reactor Coolant System total flow rate comparison~~ that the combination of ~~R~~ THERMAL POWER and Reactor Coolant System total flow rate are restored to within the regions of restricted or permissible operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

Insert ②

4.2.~~4~~ 5.2 The Reactor Coolant System 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.~~3.5~~ 5.3 The Reactor Coolant System total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

Unit 1

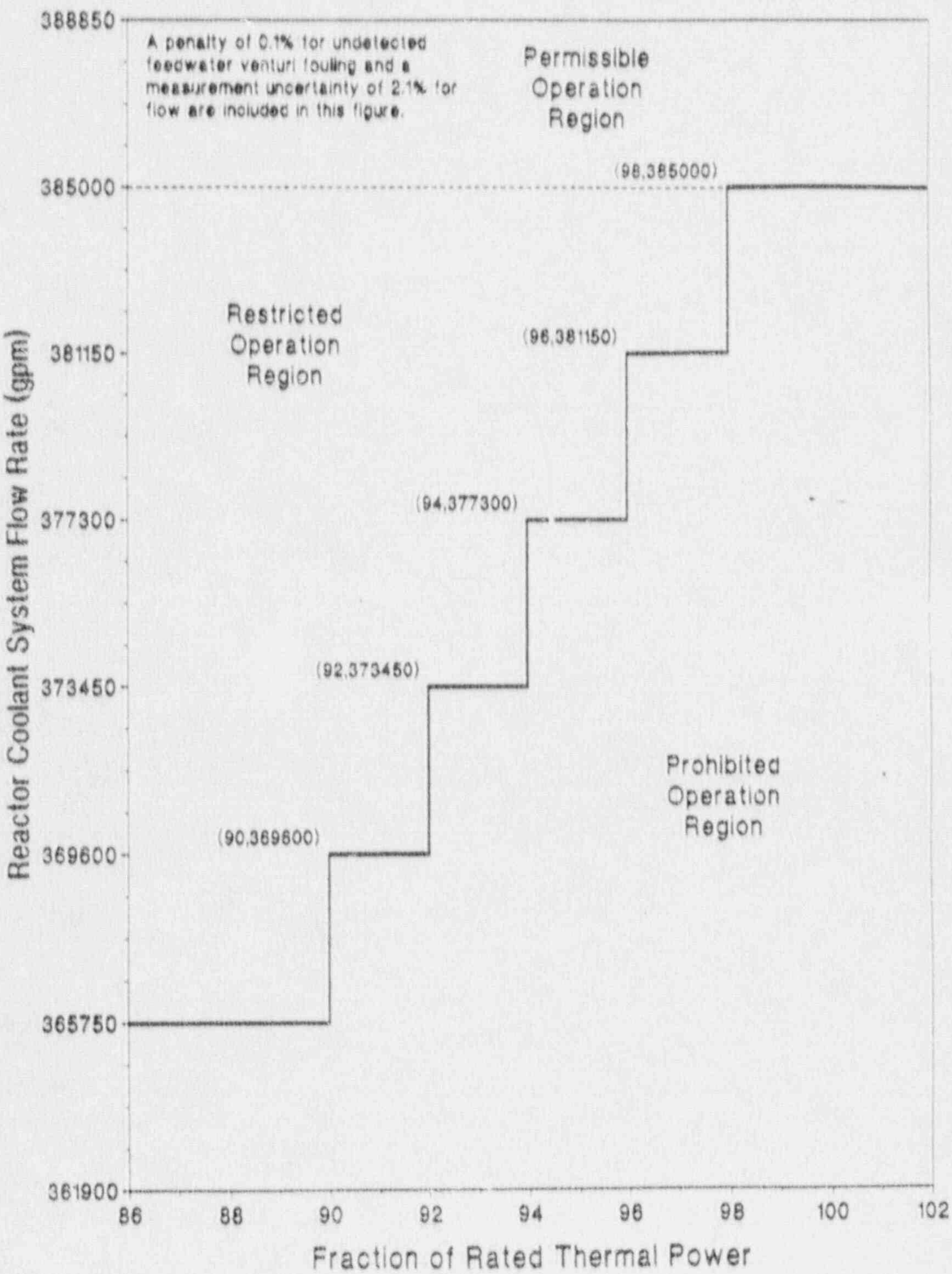
TABLE 3.2-1

DNB PARAMETERS

PARAMETER		LIMITS Four Loops in Operation
Average Temperature		
Meter Average	- 4 channels:	≤ 592°F
	- 3 channels:	≤ 592°F
Computer Average	- 4 channels:	≤ 593°F
	- 3 channels:	≤ 593°F
Pressurizer Pressure		
Meter Average	- 4 channels:	≥ 2227 psig*
	- 3 channels:	≥ 2230 psig*
Computer Average	- 4 channels:	≥ 2222 psig*
	- 3 channels:	≥ 2224 psig*
Reactor Coolant System Total Flow Rate		Figure 3.2-1

*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.

Figure 3.2-1. Reactor Coolant System Total Flow Rate Versus Rated Thermal Power - Four Loops in Operation (UNIT 1)



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Unit 2

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

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

a. the allowed operational space as specified in the CORE OPERATING LIMITS REPORT (COLR) for RAOC operation, or

b. within the target band specified in the COLR about the target flux difference during baseload operation.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER~~*~~* (Unit 2)

ACTION:

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

1. Either restore the indicated AFD to within the COLR 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 indicated AXIAL FLUX DIFFERENCE outside of the applicable target band about the target flux difference:

1. Either restore the indicated AFD to within the COLR specified target band limits 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.

---

*See Special Test Exceptions Specification 3.10.2.

**APL^{ND} is the minimum allowable (nuclear design) power level for base load operation and is specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

Unit 2

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.2.1.1 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:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

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.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least two OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, the target axial 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.

4.2.1.4 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 3/4.2.2 or by linear interpolation between the most recently measured values and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

Unit 2

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Unit 2

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Unit 2

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

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

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

Where: F_Q^{RTP} = the F_Q Limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

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

$K(Z)$ = the normalized $F_Q(Z)$ for a given core height specified in the COLR.

APPLICABILITY: MODE 1. (Unit 2)

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 (value of K_4) have been reduced at least 1% (in ΔT span) 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.

Unit 2

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 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 are satisfied.

c. Satisfying the following relationship:

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

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

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 per Specification 6.9.1.9.

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

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

*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 a power distribution map obtained.

Unit 2

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \quad \frac{F_Q^M(z)}{K(z)}$$

has increased 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 4.2.2.2c., or

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

$$\underset{\text{over } z}{\text{maximum}} \quad \frac{F_Q^M(z)}{K(z)} \quad \text{is not increasing.}$$

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

1) Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

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

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

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 AFD limits of Specification 3.2.1 by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

Unit 2

SURVEILLANCE REQUIREMENTS (Continued)

g. The limits specified in Specifications 4.2.2.2c., 4.2.2.2e., and 4.2.2.2f., 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.

4.2.2.3 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 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within the target band about the target flux difference of Specification 3.2.1) 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 FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \underset{\text{over } Z}{\text{minimum}} \left[\frac{F_Q^{RTP} \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\%$$

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. $W(z)_{BL}$ 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 CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3a shall be satisfied before re-entering Base Load operation.

4.2.2.4 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 are satisfied.

---

*APL^{ND} is the minimum allowable (nuclear design) power level for Base Load operation in Specification 3.2.1.

Unit 2

SURVEILLANCE REQUIREMENTS (Continued)

c. Satisfying the following relationship:

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

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. 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. $W(Z)_{BL}$ 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 CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

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 surveillance 4.2.2.3 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 measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)}\right]$$

has increased since the previous determination $F_Q^M(Z)$ either of the following actions shall be taken:

1. $F_Q^M(Z)$ shall be increased by 2 percent over that specified in 4.2.2.4c, or

2. $F_Q^M(Z)$ shall be measured at least once per 7 EFPD until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)}\right] \text{ is not increasing.}$$

f. With the relationship specified in 4.2.2.4c above not being satisfied, either of the following actions shall be taken:

1. Place the core in an equilibrium condition where the limit in 4.2.2.2c is satisfied, and remeasure $F_Q^M(Z)$, or

Unit 2

SURVEILLANCE REQUIREMENTS (Continued)

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated with the following expression:

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

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

1. Lower core region 0 to 15 percent, inclusive.

2. Upper core region 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 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.

Unit 2

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Unit 2

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Unit 2

3/4.2.3 REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System total flow rate and R shall be maintained within the region of permissible operation specified in the CORE OPERATING LIMITS REPORT (COLR) for four loop operation.

Where:

a. $$R = \frac{F^{N}_{\Delta H}}{F^{RTP}_{\Delta H}[1.0 + MF_{\Delta H}\ (1.0 - P)]},$$

b. $$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}},$$

c. $F^{N}_{\Delta H}$ = Measured values of $F^{N}_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^{N}_{\Delta H}$ shall be used to calculate R since the figure specified in the COLR includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 2.1% for flow and 4% for incore measurement of $F^{N}_{\Delta H}$,

d. $F^{RTP}_{\Delta H}$ = The $F^{N}_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

e. $MF_{\Delta H}$ = The power factor multiplier specified in the COLR.

APPLICABILITY: MODE 1 (Unit 2)

ACTION:

a. With the combination of Reactor Coolant System total flow rate and R within the region of restricted operation within 6 hours reduce the Power Range Neutron Flux-High Trip Setpoint to below the nominal setpoint by the same amount (% RTP) as the power reduction required by the figure specified in the COLR.

b. With the combination of Reactor Coolant System total flow rate and R within the region of prohibited operation specified in the COLR:

1. Within 2 hours either:

   a) Restore the combination of Reactor Coolant System total flow rate and R to within the region of permissible operation, or

   b) Restore the combination of Reactor Coolant System total flow rate and R to within the region of restricted operation and comply with action a. above, or

Unit 2

3/4.2.3 REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) 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.

2. Within 24 hours of initially being within the region of prohibited operation specified in the COLR, verify through incore flux mapping and Reactor Coolant System total flow rate comparison that the combination of R and Reactor Coolant System total flow rate are restored to within the regions of restricted or permissible operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours. 5

Unit 2

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

3. 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 b.1.c) and/or b.2., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated Reactor Coolant System total flow rate are demonstrated, through incore flux mapping and Reactor Coolant System total flow rate comparison, to be within the regions of restricted or permissible operation specified in the COLR prior to exceeding the following THERMAL POWER levels: 5

   a) A nominal 50% of RATED THERMAL POWER,

   b) A nominal 75% of RATED THERMAL POWER, and

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated Reactor Coolant System total flow rate determined by process computer readings or digital voltmeter measurement and R shall be determined to be within the regions of restricted or permissible operation specified in the COLR: 5

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.

4.2.3.3 The indicated Reactor Coolant System total flow rate shall be verified to be within the regions of restricted or permissible operation specified in the COLR at least once per 12 hours when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist. 5

4.2.3.4 The Reactor Coolant System 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.

4.2.3.5 The Reactor Coolant System total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

Unit 2

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Unit 2

POWER DISTRIBUTION LIMITS

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 ~~above 50% of RATED THERMAL POWER.~~

APPLICABILITY: MODE 1,*above 50% of RATED THERMAL POWER (unit 2)

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.

Unit 2

LIMITING CONDITION FOR OPERATION

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.

Unit 2

LIMITING CONDITION FOR OPERATION

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.

c. The provisions of Specification 4.0.4 are not applicable.

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.

Unit 2

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1, Unit 2 only.

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 their limits at least once per 12 hours.

Unit 2

TABLE 3.2-1

DNB PARAMETERS

PARAMETER		LIMITS Four Loops in Operation
Average Temperature		
Meter Average	- 4 channels:	$\leq$ 592°F
	- 3 channels:	$\leq$ 592°F
Computer Average	- 4 channels:	$\leq$ 593°F
	- 3 channels:	$\leq$ 593°F
Pressurizer Pressure		
Meter Average	- 4 channels:	$\geq$ 2227 psig*
	- 3 channels:	$\geq$ 2230 psig*
Computer Average	- 4 channels:	$\geq$ 2222 psig*
	- 3 channels:	$\geq$ 2224 psig*

*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.

SAFETY LIMITS AND POWER DISTRIBUTION

TS BASES MARK-UPS

2.1 SAFETY LIMITS

BASES

Insert ① for unit 1

2.1.1 REACTOR CORE (For Unit 2)

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 through the WRB-1 correlation. The WRB-1 DNB correlation 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: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95% probability with 95% confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

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

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

This curve is based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 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:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]$$

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 the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the Reactor Coolant System piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire Reactor Coolant System is hydrotested at 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

~~2.1 SAFETY LIMITS~~

~~BASES~~

(1) 2.1.1 REACTOR CORE (For Unit 1)

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 through the ~~WRB-1~~ correlation. The ~~WRB-1~~ DNB correlation 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.

BWCMV

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

and in the BWCMV DNB correlation

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

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

Attachment 1 Cont'd.

~~2.1 SAFETY LIMITS~~

~~BASES~~

~~This~~ These curves ~~is~~ are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 for Westinghouse Optimized Fuel Assemblies (OFA's) and 1.55 for the BWFC Mark-BW Fuel Assemblies 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:

$$F^N_{\Delta H} = 1.49 \; [1 + 0.3 \; (1-P)]$$ For the Westinghouse OFA's

$$F^N_{\Delta H} = 1.55 \; [1 + 0.3 \; (1-P)]$$ For the BWFC Mark-BW's

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 the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

~~2.1.2 REACTOR COOLANT SYSTEM PRESSURE~~

~~The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.~~

~~The reactor vessel, pressurizer, and the Reactor Coolant System piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.~~

~~The entire Reactor Coolant System is hydrotested at 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.~~

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 calculated DNBR in the core greater than or equal to design limit DNBR 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~~ are not exceeded.

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

$F_Q(Z)$ $F_Q(X,Y,Z)$ 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;

$F^N_{\Delta H}$ $F^N_{\Delta H}(X,Y)$ 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.

K(Z) is defined as the normalized $F_Q(X,Y,Z)$ limit for a given core height.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) ~~assure~~ ensure that the $F_Q(Z)$ $F_Q(X,Y,Z)$ ~~upper bound envelope~~ and the $F_{\Delta H}(X,Y)$ limits are ~~of the F_Q^{RTP} 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. The AFD envelope specified in the COLR has been adjusted for measurement uncertainty.

~~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.~~

Delete

At power levels below APL^{ND}, the limits on AFD are defined in the COLR, i.e., that defined by the 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 limits of which are defined in the COLR, and 2) Base Load operation, which is defined as the maintenance of the AFD within a COLR specified 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 value, the Base Load operating procedure restricts

Delete

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ~~(Continued)~~ (Unit 1)

Delete

the indicated AFD to relatively small target band and power swings (AFD target band as specified in the COLR, $APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the Units 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 one 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 ΔI power operating space (for RAOC operation), or 2) outside the allowed ΔI 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 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.

The limits on heat flux hot channel factor, ~~coolant 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 are not exceeded ~~limit~~. ~~These~~ The limits are specified in the CORE OPERATING LIMITS REPORT (COLR) per Specification 6.9.1.9.

The peaking ~~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 insure 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;

The heat flux hot channel factor and nuclear enthalpy rise hot channel factor are each

HEAT FLUX HOT CHANNEL FACTOR, ~~and REACTOR COOLANT SYSTEM FLOW RATE~~ AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued) (Unit 1)

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.

Delete
Incorporated in Bases for Specification 3.2.5

$F_{\Delta H}(X,Y)$

$F_{\Delta H}^{N}$ will be maintained within its limits provided Conditions a. through d. above are maintained. ~~As noted on the figure specified in the CORE OPERATING LIMITS REPORT (COLR), Reactor Coolant System flow rate and $F_{\Delta H}^{N}$ may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the measured $F_{\Delta H}^{N}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F_{\Delta H}^{N}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.~~

Replace with Attachment 1

R as calculated in Specification 3.2.3 and used in the figure specified in the COLR, accounts for $F_{\Delta H}^{N}$ less than or equal to the $F_{\Delta H}^{RTP}$ limit specified in the COLR. This value is used in the various accident analyses where $F_{\Delta H}^{N}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. The rod bow penalty as a function of burnup applied for $F_{\Delta H}^{N}$ is calculated with the methods described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979, and the maximum rod bow penalty is 2.7% DNBR. Since the safety analysis is performed with plant-specific safety DNBR limits compared to the design DNBR limits, there is sufficient thermal margin available to offset the rod bow penalty of 2.7% DNBR.

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) function for normal operation and the $W(Z)_{BL}$ function for Base Load Operation are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

Replace with Attachment 2

Attachment 1:

The limits on the nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, are specified in the COLR as Maximum Allowab'_ Radial Peaking limits, obtained by dividing the Maximum Allowable Total Peaking (MAP) limit by the axial peak [AXIAL(X,Y)] for location (X,Y). By definition, the Maximum Allowable Radial Peaking limits will, for Mark-BW fuel, result in a DNBR for the limiting transient that is equivalent to the DNBR calculated with a design $F_{\Delta H}(X,Y)$ value of 1.55 and a limiting reference axial power shape. The Mark-BW MAP limits may be applied to OFA fuel, provided an appropriate adjustment factor is applied to provide equivalence to a 1.49 design $F_{\Delta H}(X,Y)$ for the OFA. This is reflected in the MAP limits specified in the COLR. The relaxation of $F_{\Delta H}(X,Y)$ as a function of THERMAL POWER allows changes in the radial power for all permissible control bank insertion limits. This relaxation is implemented by the application of the following factors:

$$k = [1 + (1/RRH)(1 - P)]$$

where k = power factor multiplier applied to the MAP limits

P = THERMAL POWER / RATED THERMAL POWER

RRH is given in the COLR

Attachment 2:

$F_Q^M(X,Y,Z)$ and $F\Delta HR^M(X,Y)$ are measured periodically, and comparisons to the allowable limit are made to provide reasonable assurance that the limiting criteria will not be exceeded for operation within the Technical Specification limits of Sections 2.2 (Limiting Safety Systems Settings), 3.1.3 (Movable Control Assemblies), 3.2.1 (Axial Flux Difference), and 3.2.4 (Quadrant Power Tilt Ratio). A peaking margin calculation is performed to provide a basis for decreasing the width of the AFD and $f(\Delta I)$ limits and for reducing THERMAL POWER.

When an $F_Q^M(X,Y,Z)$ measurement is obtained in accordance with the surveillance requirements of Specification 4.2.2, no uncertainties are applied to the measured peak; the required uncertainties are included in the peaking limit. When $F_Q^M(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2, the measured peak is increased by the radial-local peaking factor to convert it to a local peak. Allowances of 5% for measurement uncertainty and 3% for manufacturing tolerances are then applied to the measured peak.

When an $F\Delta HR^M(X,Y)$ measurement is obtained, regardless of the reason, no uncertainties are applied to the measured peak; the required uncertainties are included in the peaking limit.

Delete (Incorporated in Bases for Specification 3.2.5)

HEAT FLUX HOT CHANNEL FACTOR, ~~AND REACTOR COOLANT SYSTEM FLOW RATE~~ AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued) (Unit 1)

When Reactor Coolant System flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of the figure specified in the COLR. Measurement errors of 2.1% for Reactor Coolant System total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

The measurement error for Reactor Coolant System total flow rate is based upon performing a precision heat balance and using the result to calibrate the Reactor Coolant System 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 nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in the figure specified in the COLR. Any fouling which might bias the Reactor Coolant System 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 Reactor Coolant System flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated Reactor Coolant System flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified on the figure specified in the COLR.

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.

Replace with Attachment 3

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 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 $F_Q(x,y,z)$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 2 2%.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable 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 for Power Distribution Limits Bases

Attachment 3:

A peaking increase that reflects a QUADRANT POWER TILT RATIO of 1.02 is included in the generation of the AFD limits.

QUADRANT POWER TILT RATIO (Continued) (Unit 1)

Delete

flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. The normal locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. Alternate locations are available if any of the normal locations are unavailable.

3/4.2.5 DNB PARAMETERS [Revise Text as Indicated]

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 design limit DNBR throughout each analyzed transient. ① Add Insert 1 The indicated T_{avg} value and the indicated pressurizer pressure value correspond to analytical limits of 594.8°F and 2205.3 psig respectively, with allowance for measurement uncertainty. ② Add Insert 2, Attached

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. Indication instrumentation measurement uncertainties are accounted for in the limits provided in Table 3.2-1.

① Insert 1

~~$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained.~~ As noted on ~~the figure specified in the CORE OPERATING LIMITS REPORT (COLR),~~ Figure 3.2-1 Reactor Coolant System flow rate and ~~$F^N_{\Delta H}$~~ THERMAL POWER may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the ~~measured $F^N_{\Delta H}$~~ THERMAL POWER is also low) to ensure that the calculated DNBR will not be below the design DNBR value. ~~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 relationship defined on Figure 3.2-1 remains valid as long as the limits placed on the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, in Specification 3.2.3 are maintained.

② Insert text as indicated on next page

Insert 2

Insert at ② on preceding page.

When Reactor Coolant System flow rate ~~and $F^N_{\Delta H}$ are~~ is measured, no additional allowances are necessary prior to comparison with the limits of ~~the figure specified in the COLR.~~ Figure 3.2-1 since a Measurement error~~s~~ of 2.1% for Reactor Coolant System total flow rate ~~and 4% for~~ $F^N_{\Delta H}$ ~~have~~ has been allowed for in determination of the design DNBR value.

The measurement error for Reactor Coolant System total flow rate is based upon performing a precision heat balance and using the result to calibrate the Reactor Coolant System 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 nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in ~~the figure specified in the COLR.~~ Figure 3.2-1 Any fouling which might bias the Reactor Coolant System 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 Reactor Coolant System flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

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 calculated DNBR in the core greater than or equal to design limit DNBR 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;

$F^N_{\Delta H}$ 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 (Unit 2)

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q^{RTP} 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.

POWER DISTRIBUTION LIMITS

BASES

At power levels below APL^{ND}, the limits on AFD are defined in the COLR, i.e., that defined by the 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 limits of which are defined in the COLR, and 2) Base Load operation, which is defined as the maintenance of the AFD within a COLR specified 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 value, the Base Load operating procedure restricts

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued) (Unit 2)

the indicated AFD to relatively small target band and power swings (AFD target band as specified in the COLR, $APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the Units 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 one 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 ΔI power operating space (for RAOC operation), or 2) outside the allowed ΔI 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 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.

The limits on heat flux hot channel factor, coolant 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. These limits are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

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 insure 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;

HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued) (Unit 2)

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. As noted on the figure specified in the CORE OPERATING LIMITS REPORT (COLR), Reactor Coolant System flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. 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.

R as calculated in Specification 3.2.3 and used in the figure specified in the COLR, accounts for $F^N_{\Delta H}$ less than or equal to the $F^{RTP}_{\Delta H}$ limit specified in the COLR. This value is 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. The rod bow penalty as a function of burnup applied for $F^N_{\Delta H}$ is calculated with the methods described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979, and the maximum rod bow penalty is 2.7% DNBR. Since the safety analysis is performed with plant-specific safety DNBR limits compared to the design DNBR limits, there is sufficient thermal margin available to offset the rod bow penalty of 2.7% DNBR.

The hot channel factor $F^M_Q(z)$ 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 $W(z)$ function for normal operation and the $W(Z)_{BL}$ function for Base Load Operation are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued) (Unit 2)

When Reactor Coolant System flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of the figure specified in the COLR. Measurement errors of 2.1% for Reactor Coolant System total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

The measurement error for Reactor Coolant System total flow rate is based upon performing a precision heat balance and using the result to calibrate the Reactor Coolant System 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 nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in the figure specified in the COLR. Any fouling which might bias the Reactor Coolant System 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 Reactor Coolant System flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated Reactor Coolant System flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified on the figure specified in the COLR.

3/4.2.4 QUADRANT POWER TILT RATIO (Unit 2)

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 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 movable 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

QUADRANT POWER TILT RATIO (Continued) (Unit 2)

flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. The normal locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. Alternate locations are available if any of the normal locations are unavailable.

3/4.2.5 DNB PARAMETERS (Unit 2)

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 design limit DNBR throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value correspond to analytical limits of 594.8°F and 2205.3 psig respectively, with allowance for measurement uncertainty.

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. Indication instrumentation measurement uncertainties are accounted for in the limits provided in Table 3.2-1.

ECCS TS MARK-UPS

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open,

b. A contained borated water volume of between 7853 and 8171 gallons,

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 385 and 481 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*. (UHI System Operable)

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve or boron concentration less than 1900 ppm, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

c. With one accumulator inoperable due to boron concentration less than 1900 ppm and:

   1) The volume weighted average boron concentration of the three limiting accumulators 1900 ppm or greater, restore the inoperable accumulator to OPERABLE status within 24 hours of the low boron determination 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.

   2) The volume weighted average boron concentration of the three limiting accumulators less than 1900 ppm but greater than 1500 ppm, restore the inoperable accumulator to OPERABLE status or return the volume weighted average boron concentration of the three limiting accumulators to greater than 1900 ppm and enter ACTION c.1 within 6 hours of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

*Pressurizer pressure above 1000 psig.

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LIMITING CONDITION FOR OPERATION (Continued)

3) The volume weighted average boron concentration of the three limiting accumulators 1500 ppm or less, return the volume weighted average boron concentration of the three limiting accumulators to greater than 1500 ppm and enter ACTION c.2 within 1 hour of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection 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 75 gallons by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the Reactor Coolant System pressure is above 2000 psig by verifying that power is removed from the isolation valve operators on Valves NI54A, NI65B, NI76A, and NI88B and that the respective circuit breakers are padlocked; and

d. At least once per 18 months by verifying that each cold leg injection accumulator isolation valve opens automatically under each of the following conditions:**

1) When an actual or a simulated Reactor Coolant System pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

2) Upon receipt of a Safety Injection test signal.

4.5.1.1.1.2 Each cold leg injection accumulator water level and pressure channel shall be demonstrated OPERABLE:

**This surveillance need not be performed until prior to entering HOT STANDBY following the Unit 1 refueling.

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1.2 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open,

b. A contained borated water volume of between 7704 and 8004 gallons,

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 585 and 678 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*. ~~(UHI physically disconnected; Cold Leg Accumulators and discharge paths suitably modified)~~

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve or boron concentration less than 1900 ppm, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

c. With one accumulator inoperable due to boron concentration less than 1900 ppm and:

1) The volume weighted average boron concentration of the three limiting accumulators 1900 ppm or greater, restore the inoperable accumulator to OPERABLE status within 24 hours of the low boron determination or be in at least HOT STANDBY within the next 6 hours and reduce ~~pressurizer~~ Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

2) The volume weighted average boron concentration of the three limiting accumulators less than 1900 ppm but greater than 1500 ppm, restore the inoperable accumulator to OPERABLE status or return the volume weighted average boron concentration of the three limiting accumulators to greater than 1900 ppm and

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*~~Pressurizer~~ Reactor Coolant System pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

4.5.1.1.2.2 Each cold leg injection accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

enter ACTION c.1 within 6 hours of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce ~~pressurizer~~ Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

3) The volume weighted average boron concentration of the three limiting accumulators 1500 ppm or less, return the volume weighted average boron concentration of the three limiting accumulators to greater than 1500 ppm and enter ACTION c.2 within 1 hour of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce ~~pressurizer~~ Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.2.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection 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 75 gallons by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the Reactor Coolant System pressure is above 2000 psig by verifying that power is removed from the isolation valve operators on Valves NI54A, NI65B, NI76A, and NI88B and that the respective circuit breakers are padlocked; and

d. At least once per 18 months by verifying that each cold leg injection accumulator isolation valve opens automatically under each of the following conditions:**

1) When an actual or a simulated Reactor Coolant System pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

2) Upon receipt of a Safety Injection test signal.

** This surveillance need not be performed until prior to entering HOT STANDBY following the Unit 1 refueling.

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EMERGENCY CORE COOLING SYSTEMS

UPPER HEAD INJECTION Add [Pages 3/4 5-3, 3/4 5-4 and 3/4 5-4a intentionally deleted]

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each Upper Head Injection Accumulator System shall be OPERABLE with:

a. The discharge isolation valves open,

b. A minimum contained borated water volume of 1807 cubic feet,

c. A boron concentration of between 1900 and 2100 ppm, and

d. The nitrogen-bearing accumulator pressurized to between 1185 and 1285 psig.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With the Upper Head Injection Accumulator System inoperable, except as a result of closed isolation valve(s), restore the Upper Head Injection Accumulator System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the Upper Head Injection Accumulator System inoperable due to the isolation valve(s) being closed, either immediately open the isolation valve(s) or be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each Upper Head Injection Accumulator System shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water level in the surge tank and nitrogen pressure in the accumulators, and

2) Verifying that each accumulator discharge 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 138.9 gallons by verifying the boron concentration of the solution in the water-filled accumulator;

*Pressurizer pressure above 1900 psig.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1) Verifying that each accumulator discharge isolation valve closes automatically when the water level is 93.2 ± 2.7 inches (Unit 1) and 93.1 ± 2.7 inches (Unit 2) above the working line on the water-filled accumulator, and

2) Verifying that the total dissolved nitrogen and air in the water-filled accumulator is less than 80 scf per 1800 cubic feet of water (equivalent to 5 x 10^{-5} pound of nitrogen per pound of water).

d. At least once per 5 years and if the requirements of Specification 4.5.1.2c.2) are not met by replacing the membrane installed between the water-filled and nitrogen-bearing accumulators.

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EMERGENCY CORE COOLING SYSTEMS

UPPER HEAD INJECTION

(Deleted upon the physical disconnection of the UHI System from the Reactor Coolant System)

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3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
NI-162A	Cold Leg Recirc.	Open
NI-121A	Hot Leg Recirc.	Closed
NI-152B	Hot Leg Recirc.	Closed
NI-183B	Hot Leg Recirc.	Closed
NI-173A	Residual Heat Removal Pump Disch.	Open
NI-178B	Residual Heat Removal Pump Disch.	Open
NI-100B	Safety Injection Pump Suction from Refueling Water Storage Tank	Open
NI-147B	Safety Injection Pump Mini-flow	Open

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) 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.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the residual heat removal system from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

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SURVEILLANCE REQUIREMENTS (Continued)

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 660 psig the interlocks will cause the valves to automatically close.

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 Safety Injection and Containment Sump Recirculation test signals, and

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

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) Residual heat removal pump.

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

1)	Centrifugal charging pump	$\geq$ 2380 psid,
2)	Safety Injection pump	$\geq$ 1430 psid, and
3)	Residual heat removal pump	$\geq$ 165 psid.

g. By verifying the correct position of each electrical and/or mechanical 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.

Centrifugal Charging Pump Injection Throttle Valve Number	Safety Injection Throttle Valve Number
NI-14	NI-164
NI-16	NI-166
NI-18	NI-168
NI-20	NI-170

** This surveillance need not be performed until prior to entering HOT SHUTDOWN following the Unit One first refueling.

SURVEILLANCE REQUIREMENTS (Continued)

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~~ 345 gpm, and

      b) The total pump flow rate is less than or equal to 565 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 ~~462~~ 450 gpm, and

      b) The total pump flow rate is less than or equal to 660 gpm.

   3) For residual heat removal pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3648 gpm.

ADMINISTRATIVE TS MARK-UPS

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555, with a copy to the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 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 Bank Insertion Limit for Specification 3/4.1.3.5,

3. Control Bank Insertion Limits for Specification 3/4.1.3.6,

4. Axial Flux Difference Limits, target band,* and APL^{ND}* for Specification 3/4.2.1,

5. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), W(Z),** APL^{ND}** and $W(Z)_{BL}$ for Specification 3/4.2.2, and

6. Nuclear Enthalpy Rise Hot Channel Factor, $F\Delta HR^{L****}$, or $F_{\Delta H}^{RTP****}$, and Power Factor Multiplier, $MF_{\Delta H}$,**** limits for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by 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

Insert Attachment 1

Attachment 1:

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* Reference 5 is not applicable to target band and APL^{ND}.

** References 5 and 6 are not applicable to $W(Z)$, APL^{ND}, and $W(Z)_{BL}$.

*** Reference 1 is not applicable to $F\Delta HR^{L}$.

**** Reference 5 is not applicable to $F_{\Delta H}^{RTP}$ and $MF_{\Delta H}$.

CORE OPERATING LIMITS REPORT (Continued)

Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. 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.)

3. 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.)

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

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

Insert Attachment 2

Attachment 2:

4. BAW-10152-A, "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," June 1985.
   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

5. BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse-Designed PWR's," June 1989.
   (Methodology for Specifications 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. BAW-10168P, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," September, 1989.
   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

ATTACHMENT 2

SAFETY LIMITS AND POWER DISTRIBUTION

TECHNICAL JUSTIFICATION AND

NO SIGNIFICANT HAZARDS ANALYSIS

Proposed Technical Specification Revision Figure 2.1-1

This proposed Technical Specification (TS) revision changes Figure 2.1-1 to reflect use of the BWCMV CHF correlation and B&W Fuel Company's Statistical Core Design (SCD) methodology with a 1.50 thermal design limit. The new figure applies to Unit 1 only. The figure numbers and references in T.S. 2.1.1 have been changed to indicate the correct Figure 2.1-1 for each unit.

Technical Justification

With the first batch implementation of its Mark-BW fuel design, B&W Fuel Company (BWFC) has recalculated the Catawba reactor core safety limits using its BWCMV CHF correlation along with its Statistical Core Design (SCD) methodology. With the implementation of these design methodologies for the Catawba core, it was possible to increase the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, from 1.49 to 1.55 for the Mark-BW fuel to allow greater fuel cycle design flexibility. The nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, has been maintained at 1.49 for the Westinghouse OFA's. The proposed changes to Figure 2.1-1 reflect the use of this new design limit as well as the use of BMCMV and SCD.

The reactor core safety limits provided on Figure 2.1-1 depict the combinations of thermal power, reactor coolant system pressure, and average temperature below which the calculated DNBR is no less than the design limit DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. The analysis which defined these limits was based on a full core of Mark-BW assemblies with a thermal design flow rate that bounded the minimum measured flow at Catawba. The DNB limited portions of these curves were defined using the BWCMV CHF correlation with a thermal design limit of 1.50. This 1.50 thermal design limit provided 10 percent thermal margin to the 1.345 BWCMV statistical design limit which was defined for the Catawba core using the BWFC SCD methodology. The safety limits were based on a design peaking distribution with a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine axial power shape with a peak of 1.55. To verify that this design peaking distribution was conservative on a cycle-specific basis, maximum allowable peaking (MAP) limits that provided DNB equivalence to the design distribution at various safety limit statepoints were defined. To verify that margin was available, these MAP limits were then compared to the cycle-specific peaking distribution. As part of the safety limit/MAP limit analyses, an evaluation was performed which showed that if power was reduced below 100 percent, peaking could be increased according to the following relationship:

$$k = 1 + 0.3(1 - P)$$

where k = the factor by which the MAP limits are adjusted to define reduced power limits

p = the fraction of rated power

Comparison of the Mark-BW safety limits to the Westinghouse OFA safety limits that they are replacing shows that at all points the Mark-BW limits fall outside the OFA limits. As stated in the bases for the OFA safety limits, the OFA curves were based on a nuclear enthalpy rise hot channel factor of 1.49. Mixed core studies have shown that the new Mark-BW safety limits are also applicable to the Westinghouse OFA's if the nuclear enthalpy rise hot channel factor for the OFA's is maintained at 1.49. To further ensure the applicability of the Mark-BW safety limits to the OFA fuel, the Mark-BW maximum allowable peaking (MAP) limits were adjusted at all points by the ratio of the design peaks (1.49/1.55). Mixed core studies verified that this peaking adjustment is conservative for the OFA.

<u>Proposed Technical Specification Revision Table 2.2-1</u>

In footnote to Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, loop minimum measured flow is changed from 96,900 gpm to 96,250 gpm (Unit 1 only). For Power Range Neutron Flux, High Setpoint and Low Setpoint, both Z values change from "4.56" to "5.92", where Z is the residual channel uncertainty when the calibration and drift components have been removed. The High Setpoint ALLOWABLE VALUE changes from "111.1%" to "110.9%".

The High Pressurizer Pressure Z value changes from "4.96" to "0.71" and the Sensor Error(s) from "0.5" to "1.5". The Low Reactor Coolant Flow TOTAL ALLOWANCE changes from "2.5" to "2.92", the Z value from "1.41" to "1.48", and the ALLOWABLE VALUE from "88.8%" to "88.9%". The Overtemperature ΔT Z value changes from "5.41" to "3.0", the S value from "2.65" to 2.12", and the TOTAL ALLOWANCE from "8.9" to "6.98". The table has also been updated to reflect deletion of the RTD Bypass System.

Technical Justification Table 2.2-1

The loop minimum measured flow at Catawba is contained in the footnotes of Technical Specification Table 2.1-1. Beginning with Catawba 1 Cycle 6, the value specified is being reduced from 96,900 gpm to 96,250 gpm. These values, which are one-fourth of the total design RCS flow, reflect a reduction in the nominal thermal design flow rate from 387,600 gpm to 385,000 gpm. For Catawba 1 Cycle 6, all safety and operating limit thermal-hydraulic analyses have been based on a nominal thermal design flow rate of 385,000 gpm. For SCD analyses, this value was used as is, for non-SCD analyses this value was reduced by 2.2 percent.

Changes made to the high and low power range neutron flux, high pressurizer pressure, and low RCS flow trip functions are being made to reflect both the B&W safety analysis assumptions as well as revised instrument uncertainties.

New safety analysis assumptions for the Low Reactor Coolant Flow and Overtemperature ΔT reactor trips necessitate changes to the TOTAL ALLOWANCE for these trip functions since it was desired to maintain the plant trip setpoints unchanged. Plant-specific instrument uncertainty calculations are the technical basis for the changes to the ALLOWABLE VALUE, and Z terms for all five trip functions. The ALLOWABLE VALUES for both the Power Range Neutron Flux High Setpoint and the Low Reactor Coolant Flow trips are conservatively being made more restrictive as a result of this error calculation.

S and Z are less significant parameters which may be employed to determine the operability of the channel should the trip setpoint drift beyond its ALLOWABLE VALUE. The net effect of the changes to the S and Z values for the High Pressurizer Pressure Low Reactor Coolant Flow, and Overtemperature ΔT trips is to permit a larger setpoint drift before the channel must be declared inoperable. This is, in effect, increasing the margin between the error adjusted trip setpoint and the safety analysis assumption. Conversely, the proposed changes to the Z values for the High and Low Power Range Neutron Flux setpoints conservatively restrict the permissible rack drift. Since this action statement provision has never been taken advantage of at Catawba, no past operability determination is invalidated by this change.

Since deletion of the RTD Bypass System has been completed on Catawba Units 1 and 2, Table 2.2-1 has been updated to reflect only those values which are applicable upon deletion of the RTD Bypass System as indicated by "#". This change is administrative only.

Revision to Technical Specifications Table

The $f_2(\Delta I)$ function was set to zero for the OPΔT setpoints.

Discussion

The OPΔT setpoints provide protection for Centerline Fuel Melt (CFM) as stated in BAW-10163P-A. This reference states that either $f_1(\Delta I)$ or $f_2(\Delta I)$ could be used in the safety setpoints if either $f(\Delta I)$ function protected both DNBR and CFM limits. It has been shown for Catawba 1 Cycle 6 that the OTΔT $f_1(\Delta I)$ function and the OPΔT trip function without $f_2(\Delta I)$ provide both DNBR and CFM protection. The OTΔT $f_1(\Delta I)$ function provides imbalance protection and the OPΔT provides overpower protection. The example technical specifications in BAW-10163P-A are based on $f_2(\Delta I)$ setpoints and these setpoints are adjusted when negative CFM margins are calculated. Since $f_1(\Delta I)$ provides the imbalance protection instead of $f_2(\Delta I)$, $f_1(\Delta I)$ would replace $f_2(\Delta I)$ in this specification. However, changes to the $f_1(\Delta I)$ function during operation are difficult and undesirable. Therefore, the reduction of $f_1(\Delta I)$ for negative

CFM margins was replaced by a reduction of the K_1 value in the OTΔT trip function.

Technical Justification

The Technical Specifications presented herein reduce the K_1 value of the OTΔT setpoints by an amount that is equivalent to reducing $f_1(\Delta I)$ function for negative CFM margin calculations.

Revisions To Power Distribution Specifications 3/4.2.1, 3/4.2.2, 3/4.2.3, 3/4.2.4, and 3/4.2.5

The current Power Distribution TS have been changed to reflect applicability to Unit 2 only. This has been done by placing "(Unit 2)" in the Applicability Section of the current TS. The Unit 2 Power Distribution TS will have "B" placed in front of the page number (Ex: 3/4 B2-7b), and the Unit 1 Power Distribution changes will have "A" placed in front of the page numbers. The Unit 1 TS will be changed as marked up and copied on white paper. The Unit 2 Power Distribution TS will be changed as described and copied on yellow paper.

The Power Distribution TS will be separated by Unit, and placed on different colored paper during the period of time when the Unit TS are different because of the change to Mark-BW fuel, to help ensure that the TS are applied to the correct Unit.

Revisions to Technical Specification 3/4.2.1

The target AFD for Base Load operation and the RAOC limits have been replaced with an envelope of allowed AFD values at various power levels. The AFD setpoints given in the COLR replace the RAOC operating space referred to in the current Specification. Since the reactor is not constrained to operate at a specified target AFD, the target AFD and associated band have been eliminated from Specification 3.2.1. The allowable operating AFD space is anywhere within AFD setpoint envelope in the COLR. This change applies to Unit 1 only.

Technical Specification Justification 3/4.2.1

Specification 3.2.1 was revised to provide an LCO statement and required actions consistent with BWFC methodology for core power distribution control as discussed in BAW-10163P-A (Reference 1).

During the reload licensing analyses for Catawba Unit 1, cycle 6, BWFC performed a three-dimensional maneuvering analysis to determine the Axial Flux Difference (AFD) limits based on the methodology of BAW-10163P-A. Specification 3/4.2.1 was revised for consistency with the new analytical methodology and to reflect the results of the cycle 6 analysis. The resulting AFD setpoints were placed in the Core Operating Limits Report (COLR).

The AFD limits of Specification 3.2.1 prevent the core power distribution from exceeding the allowable values based on the LOCA peaking limits and the initial condition DNB maximum allowable peaking (MAP) limits during power operation. The AFD limits are defined by a three-dimensional core maneuvering analysis that determines core peaking dependence on core loading, fuel depletion, thermal-hydraulic statepoint, control rod position, and xenon distribution. Correlations between peaking margin and axial power offset are developed that allow determination of negative and positive offset limits at selected power levels. The resulting offset limits preclude operation with negative margin, and are translated into corresponding AFD limits. The peaking margins are calculated from augmented nodal peaks calculated as described in BAW-10163P-A. The margin database comprises calculations from the entire range of power distributions generated in the maneuvering analysis, including control bank insertion to the insertion limit and transient xenon conditions.

Separate K(Z) peaking limits for Mark-BW fuel and OFA fuel were used in the maneuvering analysis to compute LOCA margins. The AFD limits determined for cycle 6 were adjusted for measurement uncertainty. The adjustment was applied at each power level between 100% and 50% of rated thermal power. The AFD setpoints given in the cycle 6 COLR are these adjusted values of the limits.

K(Z) limits for the Mark-BW fuel were determined by the ECCS analysis performed for Mark-BW fuel, as documented in BAW-10174 (Reference 2). The K(Z) limits for the OFA fuel are the current Westinghouse K(Z) values for Catawba, as given in the current Catawba COLR.

Initial condition DNB peaking margins were computed from the augmented peaks and the MAP limits based on Statistical Core Design (SCD) methodology, as described in BAW-10170P-A (Reference 3). The MAP limits are a family of peaking limits for which either the minimum DNBR is equal to the thermal design limit, or the coolant quality at the minimum DNBR location is equal to the CHF correlation quality limit. The MAP limits provide linkage between the DNBR analyses, with their design peaking distributions, and the core operating limits. The operating limit maps are based on the statepoint that represents the point of minimum DNBR during the most limiting non-OTΔT transient. To ensure applicability of the Mark-BW MAP limits to the OFA fuel, the Mark-BW MAP limits were adjusted at all points by the ratio of the design peaks (1.49/1.55). Thermal/hydraulic analyses of mixed cores verified that this peaking adjustment is conservative for OFA fuel.

References

1. BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June, 1989.

2. BAW-10174, Mark-BW Reload LOCA Analysis for Catawba and McGuire.

3. BAW-10170P-A, Statistical Core Design for Mixing Vane Cores, December, 1988.

ns to Technical Specification 3/4.2.2

This change applies to Unit 1 only.

Specification 3/4.2.2 was revised to reflect the power peaking surveillance method described in BAW-10163P-A. These revisions are summarized as follows:

1. The statement of the LCO was revised to reflect the nomenclature for the heat flux hot channel factor [($F_Q(X,Y,Z)$] used in BAW-10163P-A and throughout the Reload Report. Also, as discussed above, separate K(Z) curves are provided for the different fuel types (Mark-BW and OFA).

2. Action a in the current specification has been replaced by Actions a, b, and c in the new specification. The thermal power reduction required when $F_Q(X,Y,Z)$ exceeds its limit are the same as the current requirement, as is the reduction required in the OP T trip setpoints. Action [illegible] is a new requirement, and is provided to limit the allowable AFD when $F_Q(X,Y,Z)$ exceeds its limit. This reduces the possibility of operating the core in excess of the $F_Q(X,Y,Z)$ limit when a margin calculation (discussed in item 7 below) indicates negative operational margin exists.

3. There is no change to SR 4.2.2.1.

4. SR 4.2.2.2 addresses obtaining an incore flux map and the requirements based on the results of the measurement. The reference to RAOC operation has been deleted, since RAOC operation is unique to Westinghouse methodology.

5. There is no change to SR 4.2.2.2.a.

6. SR 4.2.2.2.b in the current surveillance has been deleted. The allowances for measurement uncertainty and manufacturing tolerances have been included in the limit [$F_Q^L(X,Y,Z)$] and therefore the measured peak $F_Q^M(X,Y,Z)$ is not increased by these factors.

7. SR 4.2.2.2.c in the current surveillance has been deleted. No simple determination is made of only whether or not the limit has been exceeded. Instead, the amount by which the 4.2.2.2 measured value is above or below the limit is qualified as detailed below.

8. SR 4.2.2.2.d (current surveillance) specifies the frequency for measuring the core power distribution. This is done by part b in the new surveillance. Part b.3 has been added to this surveillance, requiring an $F_Q(X,Y,Z)$ measurement when the excore quadrant power tilt ratio is normalized using incore detector measurements. This ensures that the impact of any core tilt on $F_Q(X,Y,Z)$ will be determined and reflected in the margin calculations of part c.

9. SR 4.2.2.2.e has been replaced by SR 4.2.2.2.d in the new surveillance. The intent of these requirements is similar in that projections of the measurements are made to determine at what point peaking would exceed allowable limits if the current trend continues. In the new surveillance, an incore flux map is obtained and the margin calculations are performed at the time when zero margin is projected. This requirement ensures the core is monitored at a frequency that considers conditions when measured peaks are underpredicted.

10. The new SR 4.2.2.2.c replaces 4.2.2.2.f in the current surveillance. The purpose of part c.1 is to perform margin calculations based on the measured peaks. With the new methodology, the limit ($[F_Q^L(X,Y,Z)]$) to which the measurement is compared is the design peak at steady-state conditions, increased by a factor that represents the maximum amount that the power at the given assembly location and axial elevation can increase above the design value before the measured value may become limiting. Margins to both the LOCA peaking limit (operational margin) and the centerline fuel melt limit (RPS margin) are calculated. The operational margin forms the basis for restricting the AFD limits in part c.2, and the RPS margin forms the basis for reducing the OPΔT trip setpoint in part c.3.

11. SR 4.2.2.2.c.2 (new) replaces SR 4.2.2.2.f.2 in the current surveillance. The reduced AFL limits determined in part c.2 are based on the amount of negative operational margin resulting from the margin calculation of part c.1. The parameters $NSLOPE_1$ and $PSLOPE_1$ are the maximum negative and positive AFD reductions required per percent margin change, and are determined from the maneuvering analysis. These parameters will be given

in the COLR. The AFD must be controlled to these new limits to reduce $F_Q(X,Y,Z)$, and to ensure that peaking will be limited for continued power operation.

12. SR 4.2.2.2.c.2.b (new) corresponds to SR 4.2.2.2.f.2.b (current surveillance).

13. Part 4.2.2.2.c.3 has been added to the surveillance. This part of the surveillance requires reducing the K_1 value of the OTΔT trip setpoint if the RPS margin is negative. This requirement ensures that centerline fuel melt protection exists when core peaking may be greater than the design values for the specified time in fuel cycle and operational conditions.

14. SR 4.2.2.2.f.2.c, which addresses Base Load operation, has been deleted from the new surveillance. The new power distribution methodology does not recognize this approach and does not constrain core operation to a target AFD.

15. SR 4.2.2.2.g has been replaced by SR 4.2.2.2.e in the new surveillance; there are no substantive changes to this surveillance.

16. SR 4.2.2.3 addresses Base Load operation and has been deleted from the new surveillance.

17. SR 4.2.2.4 addresses surveillance of peaking in Base Load operation and has been deleted from the new surveillance.

18. SR 4.2.2.5 has been replaced by SR 4.2.2.3 in the new surveillance; there are no substantive changes to this surveillance.

19. SR 4.2.2.2.1, SR 4.2.2.2.2, and SR 4.2.2.2.3 address F_{xy} monitoring and have been deleted from SR 4.2.2. The new methodology for $F_Q(X,Y,Z)$ surveillance utilizes peaking limits based on three-dimensional calculations exclusively, and does not address $F_Q(X,Y,Z)$ monitoring against planar peaking factors.

Technical Justification: 3/4.2.2

Specification 3/4.2.2 was revised to provide required actions and surveillance requirements consistent with BWFC methodology for core power distribution control and surveillance of the heat flux hot channel factor, as discussed in BAW-10163P-A (Reference 1).

The heat flux hot channel factor [$F_Q(X,Y,Z)$] is a specified acceptable fuel design limit that preserves the initial conditions for the ECCS analysis. $F_Q(X,Y,Z)$ is defined as the maximum local heat flux on the surface of a fuel rod at a given core elevation (Z) in an assembly located at (X,Y), divided by the average fuel rod heat flux, allowing for manufacturing tolerances on the fuel pellets and fuel rods. Since $F_Q(X,Y,Z)$ is a ratio of local surface heat fluxes, it is related to the total local power density in a fuel rod. Operation within the $F_Q(X,Y,Z)$ limits given in the Core Operating Limits Report (COLR) prevents power peaking that would exceed the loss of coolant accident (LOCA) peaking limits derived by the ECCS analysis. The $F_Q(X,Y,Z)$ limit is stated as the product of the peaking limit at rated thermal power (F_Q^{RTP}) and the normalized peaking limit as a function of core elevation [K(Z)]. Separate K(Z) peaking limits for Mark-BW fuel and OFA fuel were used in the maneuvering analysis to determine the operating limits. The K(Z) limits for the Mark-BW fuel were determined by the ECCS analysis performed for Mark-BW fuel, as documented in BW-10174 (Reference 2). The K(Z) limits for the OFA fuel are the current Westinghouse K(Z) values for Catawba, as given in the current Catawba COLR.

The reload maneuvering analysis determines limits on global core parameters that reflect the core power distribution. The primary parameters used to monitor and control the core power distribution are control bank insertion, axial flux difference (AFD), and quadrant power tilt ratio. Limits are placed on these parameters to ensure the core power peaking factors remain bounded during power operation. Nuclear design model calculational uncertainty, manufacturing tolerances (engineering hot channel factor), effects of fuel densification and rod bow, and modeling simplifications (such as treatment of spacer grid effects) are accommodated through the use of peaking augmentation factors in the maneuvering analysis.

Measurement of the core power distribution at steady-state conditions by using the incore detectors to obtain a three-dimensional flux map provides confirmation that the measured heat flux hot channel factor $F_Q(X,Y,Z)$ is within the values of the designed core power distribution. This comparison verifies the applicability of the design power level, control bank insertion, AFD, and excore quadrant power tilt ratio to the measured core conditions to preserve the LOCA peaking criteria.

<u>References</u>

1. BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June, 1989.

2. BAW-10174, Mark-BW Reload LOCA Analysis for Catawba and McGuire.

<u>Revisions to Technical Specification 3/4.2.3</u>

This change applies to Unit 1 only.

Specification 3/4.2.3 was revised to reflect the power peaking surveillance method described in BAW-10163P-A. These revisions are summarized as follows:

1. The statement of the LCO was revised to reflect new nomenclature for the nuclear enthalpy rise hot channel factor $[F^N_{\Delta H}(X,Y)]$ and related parameters required by the methodology of BAW-10163P-A and used throughout the Reload Report.

2. Those requirements of Actions a, b, and c in the current specification relating to the Reactor Coolant System flow rate have been incorporated in Specification 3.2.5. The Actions have been revised to include the reduction of allowable thermal power when $F\Delta HR^M(X,Y)$ exceeds the limit within 2 hours. The factor (RRH), by which the power level is decreased per percent $F_{\Delta H}(X,Y)$ is above the limit, is defined in the COLR. The inverse of this factor is the fractional increase in the MAPs allowed when thermal power is decreased by 1% RTP. When a power level decrease

is required because $F_{\Delta H}(X,Y)$ has exceeded its limit, then Action b requires restoration of $F_{\Delta H}(X,Y)$ to within its limit, or a reduction in the high flux trip setpoint. The amount of reduction of the high flux trip setpoint is governed by the same factor (RRH) that determines the thermal power level reduction. This maintains core protection and an operability margin at the reduced power level similar to that at rated thermal power.

3. Action b.3 was replaced by Action d. The portions of the Action requirements related to Reactor Coolant System flow rate have been incorporated in Specification 3.2.5, and do not appear in Action d of the new specification.

4. There is no change to SR 4.2.3.1.

5. SR 4.2.3.2 formerly covered only surveillance frequency. It has been expanded as detailed below to reflect the power peaking surveillance method described in BAW-10163P-A and the format of the revised SR 4.2.2.2. Part a addresses obtaining an incore flux map.

6. SR 4.2.3.2.b (new) replaces the current 4.2.3.2 and addresses the frequency for confirming that $F_{\Delta H}(X,Y)$ is within its limit. In addition to performing the surveillance prior to operation above 75% RTP after each fuel loading and at least once per 31 EFPD, the revised surveillance requires measurement of the peaking factor whenever the excore quadrant power tilt ratio is normalized using incore detector measurements. This ensures that the impact of any core tilt on $F_{\Delta H}(X,Y)$ will be determined and reflected in the margin calculation. This is comparable to the new SR 4.2.2.2.b on $F_Q(X,Y,Z)$.

7. SR 4.2.3.2.c has been added. The purpose of part c.1 is to perform margin calculations based on the measured radial peak ratio. The limit $[F\Delta HR^L(X,Y)]$ to which the measurement is compared is based on the allowable design MAP limit, increased by a factor that represents the maximum amount that the power at the given assembly location can increase above the design value before the measured value may become limiting.

Part c.2 uses the amount of margin determined by this procedure to form the basis for the amount of power level reduction and the reduction in the high flux and OTΔT K_1 trip setpoints required in the ACTION statements for the specification. This is comparable to the new SR 4.2.2.2.c on $F_Q(X,Y,Z)$.

8. SR 4.2.3.2.d has been added. This surveillance requires projections of the measurements to be made to determine at what point $F_{\Delta H}(X,Y)$ would exceed the allowable limit if the current trend continues. In part d.1, a penalty is applied to $F\Delta HR^M(X,Y)$ if the trend indicates that the measured peak would exceed the limiting peak within the 31 EFPD surveillance period, and the margin calculations are repeated. This provides additional margin, or a buffer, to help ensure that the peak will not exceed the limit prior to next 31 EFPD measurement interval. In part d.2, the measurement is obtained and the margin calculations are repeated so that appropriate actions can be taken before zero margin is reached. This surveillance ensures the core is monitored at a frequency that considers conditions when measured peaks are underpredicted. This is comparable to the new SR 4.2.2.2.d on $F_Q(X,Y,Z)$.

9. SR 4.2.3.3, 4.2.3.4, and 4.2.3.5 in the current specification address measurement of Reactor Coolant System flow rate. These requirements have been incorporated in Specification 3.2.5, and have been deleted from the revised requirements for SR 4.2.2.

Technical Justification: 3/4.2.3

Specification 3/4.2.3 was revised to provide required actions and surveillance requirements consistent with BWFC methodology for core power distribution control and surveillance of the nuclear enthalpy rise hot channel factor, as discussed in BAW-10163P-A (Reference 1).

The nuclear enthalpy rise hot channel factor [$F_{\Delta H}(X,Y)$] is a specified acceptable fuel design limit that preserves the initial conditions for the most limiting non-OTΔT DNB transient (i.e., primary protection against DNB is not provided by the OTΔT trip function). $F_{\Delta H}(X,Y)$ is defined as the ratio of the integral of linear power, along the rod with the highest integrated power, to the value of this integral along to the average rod. Since $F_{\Delta H}(X,Y)$ integrates the power along the length of the rod, it is related to the linear heat generation rate of the fuel rod, averaged over the length of the rod. When power distribution measurements from the incore detectors are obtained, the measured value of the assembly radial peaking factor is labeled $F_{\Delta H}^{M}(X,Y)$. The $F_{\Delta H}(X,Y)$ limits are preserved by the licensing design analysis, as described in BAW-10163P-A. Operation within the $F\Delta HR^{L}(X,Y)$ limits defined in the COLR ensures that the measured peaking will be within the design calculations. The $F\Delta HR^{L}(X,Y)$ limits are derived from the Maximum Allowable Peaking (MAP) limits specified in the Core Operating Limits Report (COLR).

The MAP limits are a family of maximum allowable total peaking curves, typically plotted as maximum allowable peak versus axial location of peak, parameterized by the axial peaking factor. The family of curves is the locus of points for which the minimum DNBR is equivalent to that calculated for the most limiting non-OTΔT transient (based on the reference design peaking). Therefore, the MAPs in the COLR are based on the statepoint that represents the point of minimum DNBR during this transient. The MAP limits provide linkage between the DNBR analyses, with their design peaking distributions, and the core operating limits.

Separate MAP limits are specified in the COLR for Mark-BW fuel and OFA fuel. The OFA MAPs were derived from the Mark-BW MAPs by adjusting the Mark-BW MAPs at all points by the ratio of the design peaks (1.49/1.55). Thermal-hydraulic analyses of mixed cores have verified that this peaking limit adjustment is conservative for OFA fuel.

The reload maneuvering analysis determines limits on global core parameters that can be measured directly. The primary parameters used to monitor and control the core power distribution are control bank insertion, axial flux difference (AFD), and quadrant power tilt ratio. Limits are placed on these parameters to ensure the core power peaking factors remain bounded during power operation. Uncertainties for the nuclear design model, engineering hot channel factor, assembly spacing, axial peaking factor, and other uncertainties in the CHF correlation were statistically combined to produce an overall DNBR uncertainty. This overall uncertainty was used to establish the statistical design limit (SDL), as described in BAW-10170P-A (Reference 2). Since the MAP limits link the peaking limits to the DNBR limit, these uncertainties are accounted for in the MAP limits, and are not applied explicitly in the maneuvering analysis.

Comparisons of the measured core power distribution at steady-state conditions to the design power distribution provide confirmation that the measured $F_{\Delta H}(X,Y)$ is within the values of the designed core power distribution. This comparison verifies the applicability of the measured core condition to the designed condition, so that if the control bank insertion, AFD, and excore quadrant power tilt ratio are at their most limiting values, then the initial condition DNB peaking criteria are preserved.

When the measurement is obtained, values of measured $F_{\Delta H}(X,Y)$ are not compared directly to the MAP limits. Instead, the ratio of the measured $F_{\Delta H}(X,Y)$ to the maximum allowable radial peak derived from the MAP limits is formed. The maximum allowable radial peak ratio is derived from the MAP limits by dividing the MAP for the assembly under surveillance by the measured axial peak for that assembly, and is denoted by $F\Delta HR^{M}(X,Y)$ in the COLR. The corresponding maximum allowable radial peak limit ratio is derived by dividing the MAP for the assembly by the axial peak from the design power distribution database for the particular conditions at which the measurement is made. This ratio is then increased by a factor that represents the maximum amount that the power at the given assembly location can increase above the design value before the measured value may become limiting. The resulting value is denoted by $F\Delta HR^{L}(X,Y)$ in the COLR. If $F\Delta HR^{M}(X,Y)$ is less than $F\Delta HR^{L}(X,Y)$, then a positive margin will exist, and it is inferred that $F_{\Delta H}(X,Y)$ is within the limit.

References

1. BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June, 1989.

2. BAW-10170-A, Statistical Core Design for Mixing Vane Cores, December, 1988.

Revisions to Technical Specification 3/4.2.4

Specification 3.2.4 was revised to reflect the requirement to decrease thermal power by at least 3% for each 1% of indicated quadrant power tilt ratio in excess of 1.02. This was done because the power distribution analysis includes a peaking allowance for quadrant power tilt ratios up to 1.02. When the quadrant power tilt ratio increases above 1.02, reductions in thermal power are required to limit the maximum local linear heat rate. The actions required to reduce thermal power are provided in the current specification. Therefore, this change in the specification reflects a quadrant power tilt ratio of 1.02 as the "reference" value, above which a thermal power reduction is required. This revision is consistent with the peaking allowance for quadrant power tilt, as described above. This revision applies to Unit 1 only.

The Applicability of Specification 3.2.4 is for Mode 1 operation above 50% of rated thermal power. The phrase "above 50% of rated thermal power" was removed from the LCO statement and written into the Applicability statement for clarity and consistency with the format of the Standard Technical Specifications. A statement that "The provisions of 3.0.4 are not applicable" was also added to clarify that the surveillance requirement would be completed above 50% of RATED THERMAL POWER. This change is administrative in nature because it does not represent an actual change to the requirements of Specification 3.2.4 or to its required actions. The revision to the Applicability Section applies to Unit 1 and Unit 2.

A footnote was added to the Applicability statement to indicate that the specification is not applicable until completion of excore detector calibration subsequent to refueling.

Technical Specification Justification: 3/4.2.4

Specification 3.2.4 was revised to provide required actions consistent with BWFC methodology for core power distribution control as discussed in BAW-10163P-A (Reference 1).

During the reload licensing analyses for Catawba Unit 1, cycle 6, a three-dimensional maneuvering analysis was performed to determine the core operating limits for power distribution, based on the methodology of BAW-10163P-A. The analysis addressed the Axial Flux Difference (AFD) limits (Specification 3.2.1), control bank insertion limits (Specification 3.1.3.6), and Quadrant Power Tilt Ratio (Specification 3.2.4). The current control bank insertion limits were verified by the analysis, and revised AFD limits were set based on the new power distribution methodology and the cycle 6 core design. These results include an allowance for the limiting quadrant power tilt in the core.

When the control bank insertion and AFD limits are determined during the maneuvering analysis, the calculated peaking is increased by an amount corresponding to a 2% quadrant power tilt, equivalent to a quadrant power tilt ratio of 1.02. Therefore, the resulting control bank insertion and AFD limits are valid for excore quadrant power tilt ratios up to the Technical Specification value of 1.02. Simulations of quadrant power tilts and corresponding peaking increases have shown that a peaking allowance of 3% in the analysis is sufficient to bound the increased peaking due to tilt up to a quadrant power tilt ratio of 1.02. This peaking allowance has been used in the maneuvering analysis to determine both AFD (operational) limits and f(ΔI) (safety) limits.

References

1. BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June, 1989.

Proposed Technical Specification Revision 3/4.2.5

This change applies to Unit 1 only.

This proposed Technical Specification (TS) revision changes TS 3.2.5 to include a new figure that defines power reduction requirements for low flow operation. This new figure, Figure 3.2-1, replaces Figure 3.2-3 which had previously been relocated from Specification 3.2.3 to Figure 8 of the COLR.

Technical Justification 3/4.2.5

Since maximum allowable peaking limits are still defined in Specification 3.2.3, the peaking dependence that was previously included on COLR Figure 8 in the parameter R has been eliminated from the new figure. Specification 3.2.5 has also been revised to include the action statements from Specification 3.2.3 that govern the response to the power and flow combination being in the regions of restricted or prohibited operation, while maintaining the current action statement that governs the response to temperature and pressure exceeding their limits. In addition, all flow rate surveillance requirements previously contained in Specification 4.2.3 have been moved to Specification 4.2.5.

Although Figure 3.2-3 had previously been moved to the Core Operating Limits Report (COLR), the new figure, Figure 3.2-1, is being returned to the plant Technical Specifications, since the parameters that are governed by the figure (power and flow) do not change on a cycle by cycle basis.
Previously, Figure 3.2-3 in Specification 3.2.3 related minimum flow as a function of both power and R, where R was a function of the Nuclear Enthalpy Rise Hot Channel factor, power, and two cycle specific parameters (the design

Nuclear Enthalpy Rise Hot Channel factor and the low power peaking adjustment factor). Because Figure 3.2-3 contained cycle specific parameters, it was moved from Specification 3.2.3 to the COLR. The new figure, Figure 3.2-1, relates minimum flow only to power. The dependence of minimum flow on R has been eliminated. Therefore, since the dependence of the figure on cycle specific parameters no longer exists, it can be returned to the Technical Specifications.

Figure 3.2-1 defines the trade-off in power and flow that will maintain the bases of the core safety and operating limits for a low flow condition. The analysis that verifies this trade-off considered a flow reduction down to 95 percent of the thermal design flow rate. To verify the validity of the trade-off, several DNBR evaluations were performed. These evaluations demonstrate that the Overtemperature ΔT safety limits remain valid as flow is reduced and that the specified reduction in operating power level produces improved DNBR margins for the limiting conditions II event. All reduced power statepoints considered the increase in allowable core peaking at reduced power consistent with Technical Specification 3.2.3. Therefore, no additional limits on the maximum Nuclear Enthalpy Rise Hot Channel Factor are required due to reduced flow conditions, thereby removing the R dependence.

To ensure that the level of protection that has been assumed in the plant safety analysis is maintained, all action statements that were previously included in Specification 3.2.3 have been retained in the new Specification 3.2.5. This assures that non-loss-of-flow transients, like the rod withdrawal at power, are protected at the low flow condition.

NO SIGNIFICANT HAZARDS ANALYSIS FOR SAFETY LIMITS AND POWER DISTRIBUTION TECHNICAL SPECIFICATIONS

The following analysis, required by 10CFR 50.91, concludes that the proposed amendment will not involve significant hazards considerations as defined by 10 CFR 50.92.

10 CFR 50.92 states that a proposed amendment involves no significant hazards considerations if operation in accordance with the proposed amendment would not:

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

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

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

The fuel for Catawba Nuclear Station Cycles 1-5, for both Units 1 and 2, is Westinghouse supplied. As a result of Duke Power's decision to open future reload contracts to competitive bidding, the fuel for at least Cycles 6-9 of Catawba Unit 1 and 6 and 7 of Unit 2 will be supplied by B&W Fuel Company. Unit 1 Cycle 6 will be the first cycle for which BWFC supplies the reload fuel. The Catawba Unit 1, Cycle 6 Reload Safety Evaluation Report (Attachment 3) presents an evaluation which demonstrates that the core reload using Mark-BW fuel will not adversely impact the safety of the plant. During Cycle 6 the core will contain 72 fresh fuel assemblies supplied by B&W and 121 Westinghouse supplied Optimized Fuel Assemblies (OFA). Methods and models have been developed to support Catawba Unit 1 operation during both normal and off normal operation. These methods and models ensure safe operation with an entire core of Mark-BW fuel and with a core of mixed Westinghouse and Mark-BW fuel. The analysis methods are documented in Topical Reports which have been submitted to the NRC, and are either under review or approved. These Topical Reports are listed in Section 10 of Attachment 3.

For the reload-related Technical Specifications the probability or consequences of an accident previously evaluated is not significantly increased.

A LOCA evaluation for operation of Catawba Nuclear Station with Mark-BW fuel has been completed (BAW 10174, Mark-BW Reload LOCA Analysis for the Catawba and McGuire Units). Operation of the station while in transition from Westinghouse supplied OFA fuel to B&W supplied Mark-BW fuel is also justified in this topical.

BAW 10174 demonstrates that Catawba Nuclear Station continues to meet the criteria of 10 CFR 50.46 when operated with Mark-BW fuel.

Large Break LOCA calculations completed consistent with an approved evaluation model (BAW 10168P and revisions) demonstrate compliance with 10 CFR 50.46 for breaks up to and including the double ended severance of the largest primary coolant pipe. The small break LOCA calculations used to license the plant during previous fuel cycles are shown to be bounding with respect to the new the fuel design. This demonstrates that the plant meets 10 CFR 50.46 criteria when the core is loaded with Mark-BW fuel.

During the transition from Westinghouse OFA fuel to Mark-BW fuel both types of fuel assemblies will reside in the core for several fuel cycles. Appendix A to BAW-10174 demonstrates that results presented above apply to the Mark-BW fuel in the transition core, and that insertion of the Mark-BW fuel will not have an adverse impact on the cooling of the Westinghouse fuel assemblies.

BAW 10173P, Mark-BW Reload Safety Analysis for the Catawba and McGuire Units, provides evaluations and analyses for non-LOCA transients which are applicable to Catawba. The scope of BAW 10173P includes all events specified by sections 15.1-15.6 of Regulatory Guide 1.70 (Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants) and presented in the Final Safety Analysis Report for Catawba. The analysis and evaluations performed for BAW 10173P confirm that operation for Catawba Nuclear Station for reload cycles with Mark-BW fuel will continue to be within the previously reviewed and licensed safety limits.

One of the primary objectives of the Mark-BW replacement fuel is compatibility with the resident Westinghouse fuel assemblies. The description of the Mark-BW fuel design, and the thermal-hydraulics and core physics performance evaluation demonstrate the similarity between the reload fuel and the resident fuel. The extensive testing and analysis summarized in BAW 10173P shows that the Mark-BW fuel design performs, from the standpoint of neutronics and thermal-hydraulics, within the bounds and limiting design criteria applied to resident Westinghouse fuel for the Catawba plant safety analysis.

Each FSAR accident has been evaluated to determine the effects of Cycle 6 operation and to ensure that the radiological consequences of hypothetical accidents are within applicable regulatory guidelines, and do not adversely affect the health and safety of the public. The design basis LOCA evaluations assessed the radiological impact of differences between the Mark-BW fuel and Westinghouse OFA fuel fission product core inventories. Also, the dose calculation effects from non-LOCA transients reanalyzed by BWFC utilizing Cycle 6 characteristics were evaluated. Differences in the current FSAR dose values that are not related to the insertion of Mark-BW fuel reflect the application of the latest revisions to Standard Review Plan dose assessment methodology. The calculated radiological consequences are all within specified regulatory guidelines and contain significant levels of margin.

The analyses contained in the referenced Topical Reports indicate that the existing design criteria will continue to be met. Therefore, these TS changes will not increase the probability or consequences of an accident previously evaluated.

As stated in the above discussion, normal operational conditions and all fuel-related transients have been evaluated for the use of Mark-BW fuel at Catawba Nuclear Station. Testing and analysis was also completed to ensure that from the standpoint of neutronics and thermal-hydraulics the Mark-BW fuel would perform within the limiting design criteria. Because the Mark-BW fuel performs within the previously licensed safety limits, the possibility of a new or different accident from any previously evaluated is not created.

The safety analyses performed in support of any reload necessarily involve the assumption of a number of input parameter values. Because of the differences in methodologies between vendors, and the proprietary nature of the analyses, a side-by-side comparison of input assumptions is generally neither possible nor useful. Reactor Coolant System flow is an exception, because it is a TS constrained and measurable value. The B&W analyses referenced in the above discussion assumed an RCS flow of 385,000 gpm. TS have been changed to reflect this new value (in Table 2.2-1, footnote, loop minimum measured flow = 96,250 gpm, and Figure 3.2-1, Rated Thermal Power vs. Flow). The change also affects Figure 2.1-1, Reactor Core Safety Limits. Because the new safety limits continue to provide assurance that DNB, and hot leg boiling will not occur, this change does not represent a significant decrease in the margin of safety.

The reload-related changes to the TS do not involve a significant reduction in the margin of safety. The calculations and evaluations documented in BAW 10174 show that Catawba will continue to meet the criteria of 10 CFR 50.46 when operated with Mark-BW fuel. The evaluation of non-LOCA transients documented in BAW-10173P also confirms that Catawba will continue to operate within previously reviewed and licensed safety limits. Because of this, the TS changes to support the use of Mark-BW fuel will not involve a significant reduction in the margin of safety.

Several changes have been made to Table 2.2-1. These changes reflect updated plant specific instrument uncertainty calculations. The allowable values for both the power range neutron flux high setpoint and the low RCS flow trips are conservatively being made more restrictive as a result of this error calculation. The S and Z terms are used to determine the operability of a channel if the trip setpoint exceeds its allowable value. The modifications to the S and Z values for high pressurizer pressure permit a larger rack drift before the channel must be declared inoperable. The changes to the S and Z values for the high and low neutron flux trips conservatively restrict the rack drift.

The changes to Table 2.2-1 will not significantly increase the probability or consequences of an accident previously evaluated. The changes to the allowable values for the power range neutron flux (high setpoint) and low RCS flow, and the S and Z values for power range neutron flux (both setpoints), are conservative. The modification to the Z value permits a larger rack drift for pressurizer pressure, low RCS flow, and overtemperature T before the channel becomes inoperable, however these changes more accurately represents expected values, and are within the safety analysis assumptions. For similar reasons it can be concluded that these charges will not create the possibility of any new accident from those previously evaluated. It can also be concluded that since all new TS values are bounded by safety analysis assumptions that this change will not significantly decrease the margin of safety.

Several of the requested amendments are administrative in nature. The requested change which updates Table 2.2-1 for deletion of the RTD Bypass System, reflects a change which has been previously approved by the NRC (Amendment No. 40 to Facility Operating License NPF-35 and Amendment No. 33 to Facility Operating License NPF-52). Since the needed modifications have been completed on both Catawba Units 1 and 2 the TSs which no longer apply are being deleted. Since there is no change in requirements this change does not involve significant hazards considerations.

An administrative change has been requested for TS 3.2.4 to delete "above 50% of RATEL THERMAL POWER" from the LCO, and add it to the Applicability section. A statement that "The provisions of 3.0.4 are not applicable" was also added which would clarify that the surveillance requirement would be completed above 50% RATED THERMAL POWER. This change is consistent with both the Westinghouse Standard Technical Specifications and the way the plant is currently operated. Since there is no change to the current requirements this change is administrative in nature, and involves no significant hazards considerations.

An administrative change is being made to the TS which apply to Unit 2, and no longer apply to Unit 1 after the reload. The current Figure 2.1-1 has been relabeled as applicable to Unit 2 only. Table 2.2-1 also notes that the the existing Reactor Coolant flow applies only to Unit 2 and the new Reactor Coolant flow (96,250) applies to Unit 1. The Applicability Section of the Power Distribution TS (3/4.2.1, 3/4.2.2, 3/4.2.3, 3/4.2.4) have also been revised to show that the existing TS still apply to Unit 2. The existing TS will be copied on yellow paper to further distinguish them from the new TS which apply to Unit 1 only. The Power Distribution TS will have an "A" in the page number for Unit 1 and a "B" for Unit 2, the pages will also be marked "Unit 1" or "Unit 2". This change is administrative only, and is being made to distinguish between the TS for Unit 1, which will be operated with TS revisions which reflect the use of Mark-BW fuel, and Unit 2 which will continue to operate with Westinghouse supplied fuel.

ENVIRONMENTAL IMPACT STATEMENT

The proposed TS change has been reviewed against the criteria of 10 CFR 51.22(c)(9) for environmental considerations. As shown above, the proposed change does not involve any significant hazards consideration, nor significantly increase the types or amounts of effluents that may be released offsite, nor significantly increase the individual or cumulative occupational radiation exposure. Based on this, the proposed Technical Specification change meets the criteria given in 10 CFR 51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.

TECHNICAL JUSTIFICATION AND NO SIGNIFICANT

HAZARDS ANALYSIS FOR ECCS TS

Proposed Technical Specification Revision 3/4.5

This change applies to both Unit 1 and Unit 2. This proposed TS revision deletes LCO 3.5.1.1.1 and the associated surveillance requirements. The note in the Applicability Section of 3.5.1.1.2, (UHI physically disconnected; Cold Leg accumulators and discharge paths suitably modified), is deleted. TS 3.5.1.2 and its associated surveillance requirements is also deleted.

Technical Justification 3/4.5

This change to the Technical Specifications is administrative in nature. A previously approved change to the TS allows operation of both Catawba Units 1 and 2 with the Upper Head Injection System removed. TS 3.5.1.1.1 and 3.5.1.2 along with the associated surveillance requirements no longer apply since the Upper Head Injection system has been removed. The note in the Applicability Section of 3.5.1.1.2 is no longer needed. The modification of the Upper Head Injection System has been completed on both Catawba Units. The note was included as an interim measure while the modification of the system was in progress.

Since UHI has been removed, deletion of the Specifications, which call for the system to be operable will add clarity to the TS.

Changing "pressurizer" to "Reactor Coolant System" in TS 3.5.1.1.2 ACTION c.1, 2, and 3 is a change to reflect the instrument used by the plant to complete the required ACTIONS. Pressurizer pressure goes off scale low at 1700 psig so it can not be used to measure pressure below 1000 psig as stated in the current TS.

Proposed Technical Specification Revision 4.5.2.h

This proposed amendment changes the 4.5.2.h.1.a flow rate from "333 gpm" to "345 gpm", and 4.5.2.h.2.a flow rate from "462 gpm" to "450 gpm". This change applies to Unit 1 and Unit 2. There are also administrative changes to remove items in TS 3/4.5.1 and 3.5.1.2 to reflect the fact that UHI has been physically disconnected, and to change "pressurizer" pressure to "Reactor Coolant System" pressure in Action C.1, C.2, and C.3 to TS 3.5.1.1.2.

Technical Justification 4.5.2.h

Technical Specifications 4.5.2.h.1.a and 4.5.2.h.2.a are flow requirements for the centrifugal charging pumps and safety injection pumps, respectively. Each flow value represents the minimum acceptable total injected flow through three injection lines with the Reactor Coolant System at atmospheric pressure. The flow through the fourth injection line is assumed to spill out of a pressure boundary rupture during a LOCA and is not available for core cooling. The 333 gpm value must be increased to 345 gpm in order to match the flow assumed in the new FSAR Chapter 15.6.5 LOCA analysis associated with the Catawba 1 Cycle 6 reload. The flow assumed in the existing LOCA analysis is 308 gpm. The 345 gpm value is only a small increase in the previous value and is not associated with a change in the centrifugal charging pumps or piping system. The present configuration of the system meets the 345 gpm flow requirement. 345 gpm was assumed in the new LOCA analysis since the analysis is valid for both Catawba and McGuire, and the existing McGuire Technical Specification value is 345 gpm. The proposed change from 462 to 450 gpm for the safety injection pumps is necessary to provide margin for instrument string uncertainty during flow testing, and for a reasonable tolerance on the test acceptance criteria for injected flow imbalance between the four injection lines. The flow assumed in the new LOCA analysis associated with the Catawba 1 Cycle 6 reload is 405 gpm. The flow assumed in the existi[illegible] [illegible]CA analysis is 462 gpm. The impact of decreasing the flow assumed in the existing LOCA analysis from 462 gpm to 450 gpm has been determined to be acceptable based on the excess flow available from the centrifugal charging pumps (308 gpm assumed and 345 gpm available). The flows delivered by the ECCS sub-systems can be traded off in this manner as long as the total ECCS flow assumed in the LOCA analysis is not invalidated. The proposed Technical Specifications are consistent with both the existing LOCA analysis, and the new LOCA analysis associated with the Catawba 1 Cycle 6 reload, and are therefore acceptable.

The following changes requested are administrative in nature. The deletion of Specifications in Section 3/4.5.1 which require the UHI System to be operable in the applicability, and 3.5.1.2 which is marked in the TS to be deleted when UHI is physically disconnected from the Reactor Coolant System, also reflects a change which was previously approved by the NRC (Amendent No. 32 to Facility Operating License NPF-32 and No. 23 to Facility Operating License NPF-52. Since the needed modifications have been completed on both Catawba Units 1 and 2 the TSs which no longer apply are being deleted. Another administrative change is changing "pressurizer pressure" to "Reactor Coolant System" pressure in ACTION C.1., C.2, and C.3 to TS 3.5.1.1.2. This change is administrative because it reflects the instrument used by the plant to complete the required ACTIONS. Since pressurizer pressure goes off scale low at 1700 psig, it cannot be used to measure pressure below 1000 psig as stated in the current TS. Since there is no change in requirements, this change does not involve significant hazards considerations.

NO SIGNIFICANT HAZARDS ANALYSIS
TS 3/4.5 EMERGENCY CORE COOLING SYSTEMS

The proposed changes to TS 3/4.5 (Emergency Core Cooling Systems) consist of an administrative change to remove TS related to the Upper Head Injection System and revisions to the required flowrates for the centrifugal charging pumps (TS 4.5.2.h.1.a. "333 gpm to "345 gpm") and safety injection pumps(TS 4.3.2.h.2.a "462 gpm" to "450 gpm"). The following analysis required by 10CFR 50.91 concludes that the proposed amendment does not involve a significant hazards consideration as defined by 10CFR 50.92.

10 CFR 50.92 states that a proposed amendment involves no significant hazards considerations if operation in accordance with the proposed amendment would not:

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

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

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

The proposed revisions to the required flowrates for the centrifugal charging pumps and safety injection pumps do not involve a significant increase in the probability or consequences of an accident previously evaluated. The new pump flowrates represent a change in the assumptions made in the LOCA analysis, not a physical change in the plant. The increase in the required centrifugal charging pump flows is small, and both Catawba Units currently meet the new requirement. The required flowrate for the safety injection pumps has been lowered to allow for instrument uncertainty and to allow for a reasonable tolerance in the acceptance criteria for injected flow imbalance between the four injection lines. Lowering the safety injection pump flows will be acceptable with respect to the existing LOCA analysis, because as discussed above there is excess flow available from the centrifugal charging pumps. Since a total ECCS flow value is assumed in the LOCA analysis, lowering the required flow for the safety injection pumps is acceptable for the current LOCA analysis, as long as the total ECCS flow assumed in the LOCA analysis remains available. Because the new TS requirements are consistent with both the new and the existing LOCA analyses, neither the probability or the consequences of an accident previously evaluated will be significantly increased.

The proposed Technical Specifications meet the criteria of both the new and the existing LOCA analysis. No changes have been made in the plant, and both Catawba units are currently operating within the proposed TS. Since a total ECCS flow is assumed in the LOCA analysis increasing the required centrifugal charging pump flow to account for a decrease in required safety injection flow insures that the existing LOCA analysis remains valid with the new TS requirements. Because the new TS values ensure that both the new and existing LOCA analysis remain valid, this change will not create the possibility of a new or different accident from any previously evaluated.

The LOCA analysis assumes a minimum ECCS flow. Both the new and the existing LOCA analyses remain valid with the proposed TS changes. Because the LOCA analysis remains valid, this change will not involve a significant reduction in the margin of safety.

The following changes are administrative in nature. The deletion of Specifications in Section 3/4.5.1 which require the UHI System to be operable in the applicability, and 3.5.1.2 which is marked in the TS to be deleted when UHI is physically disconnected from the Reactor Coolant System, also reflects a change which was previously approved by the NRC (Amendment No. 32 to Facility Operating License NPF-32 and No. 23 to Facility Operating License NPF-52). Since the needed modifications have been completed on both Catawba Units 1 and 2 the TSs which no longer apply are being deleted. Another administrative change is changing "pressurizer" pressure to "Reactor Coolant System" pressure in ACTION C.1, C.2, and C.3 to TS 3.5.1.1.2. This change is administrative because it reflects the instrument used by the plant to complete the required ACTIONS. Since Pressurizer pressure goes off scale low at 1700 psig, it cannot be used to measure pressure below 1000 psig as stated in the current TS. Since there is no change in requirements, this change does not involve significant hazards considerations.

ENVIRONMENTAL IMPACT STATEMENT

The proposed TS change has been reviewed against the criteria of 10 CFR 51.22(c)(9) for environmental considerations. As shown above, the proposed change does not involve any significant hazards consideration, nor significantly increase the types or amounts of effluents that may be released offsite, nor significantly increase the individual or cumulative occupational radiation exposure. Based on this, the proposed Technical Specification change meets the criteria given in 10 CFR 51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.,

NO SIGNIFICANT HAZARDS ANALYSIS

FOR TS 6.9.1.9

NO SIGNIFICANT HAZARDS ANALYSIS
ADMINISTRATIVE CHANGES TO TS 6.9.1.9

There has been an administrative change proposed to TS 6.9.1.9 (Core Operating Limits Report) to reflect the use of BWFC methodology and analyses. Note "*", "**", and "****" on Attachment 1 for Specification 6.9.1.9 are added to clarify that only References 1, 2, and 3, which reference Westinghouse methodology and analysis apply. Note "***" clarifies that Westinghouse analyses does not apply to $F\Delta HR^{L}$". The references added as Attachment 2 to TS 6.9.1.9 (BAW-10152-A "NOODLE-A multi-dimensional Two Group Simulator," BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse Designed PWR's", and BAW-10168P, Rev. 1 "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Stearn Generator Plants") reflect the use of BWFC methodology to determine the cycle specific limits in tha COLR. These references reflect methodology which is approved or under review by the NRC staff.

10 CFR 50.92 states that a proposed amendment involves no significant hazards considerations if operation in accordance with the proposed amendment would not:

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

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

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

This change is administrative in nature, reflecting the use of NRC approved methodology to determine the operating limits in the COLR. The use of BWFC methodology and analysis has been previously justified in this submittal. The administrative change to the references for the COLR will not increase the probability or consequences of an accident previously evaluated. For the above reasons it can also be concluded that this change will not create the possibility of a new or different accident from any previously evaluated.

The use of BWFC methodology and analyses have been previously determined to be acceptable by the NRC, and their use to determine the operating limits for Catawba, Unit 1, Cycle 6, is previously justified in this submittal. Because this change is administrative, simply listing the methodologies already determined to be acceptable, it does not involve a significant reduction in the margin of safety.

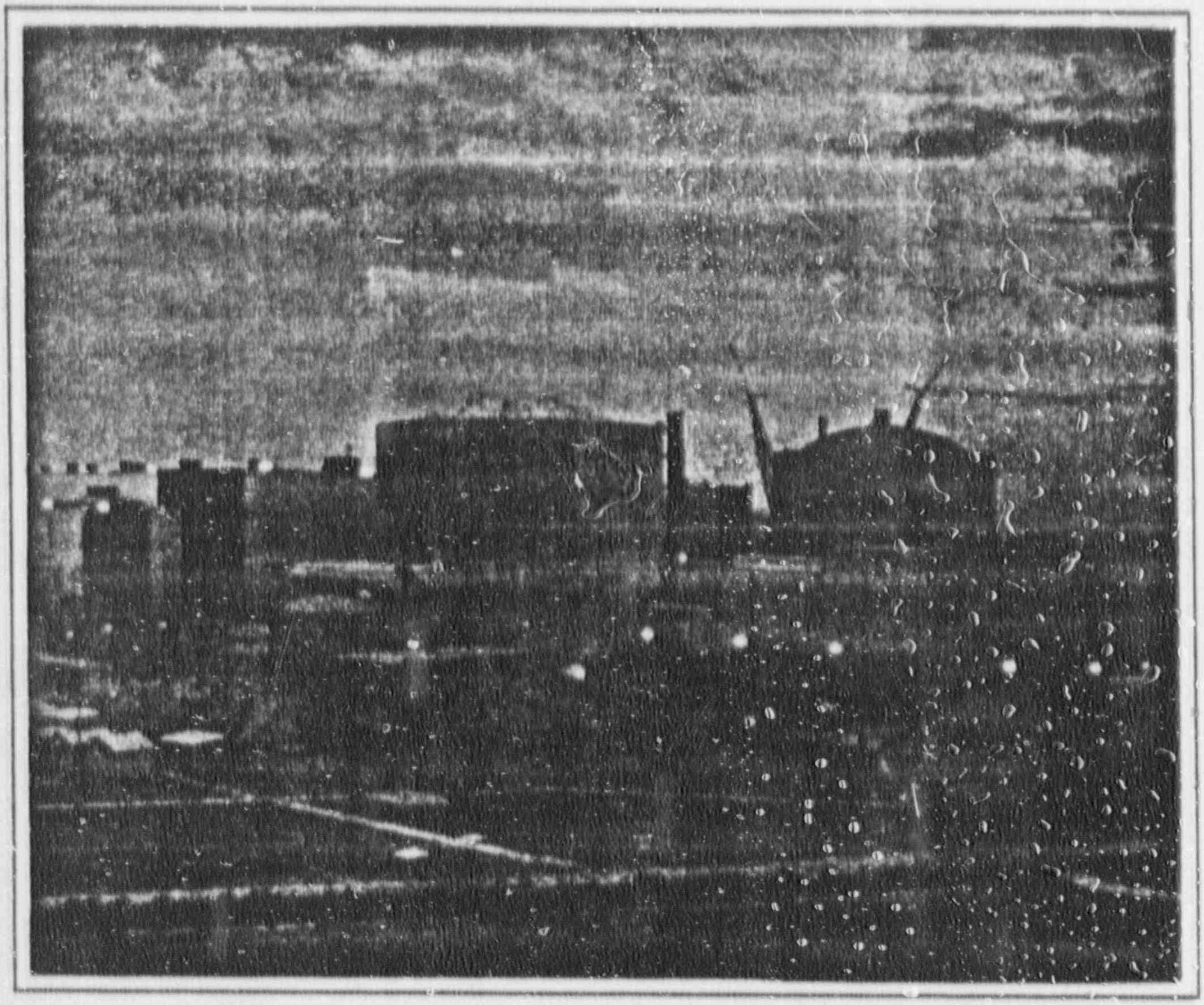
ATTACHMENT 3

BAW-2119

October 1990

RELOAD REPORT

Catawba Unit 1, Cycle 6



DUKE POWER COMPANY

Catawba Nuclear Station

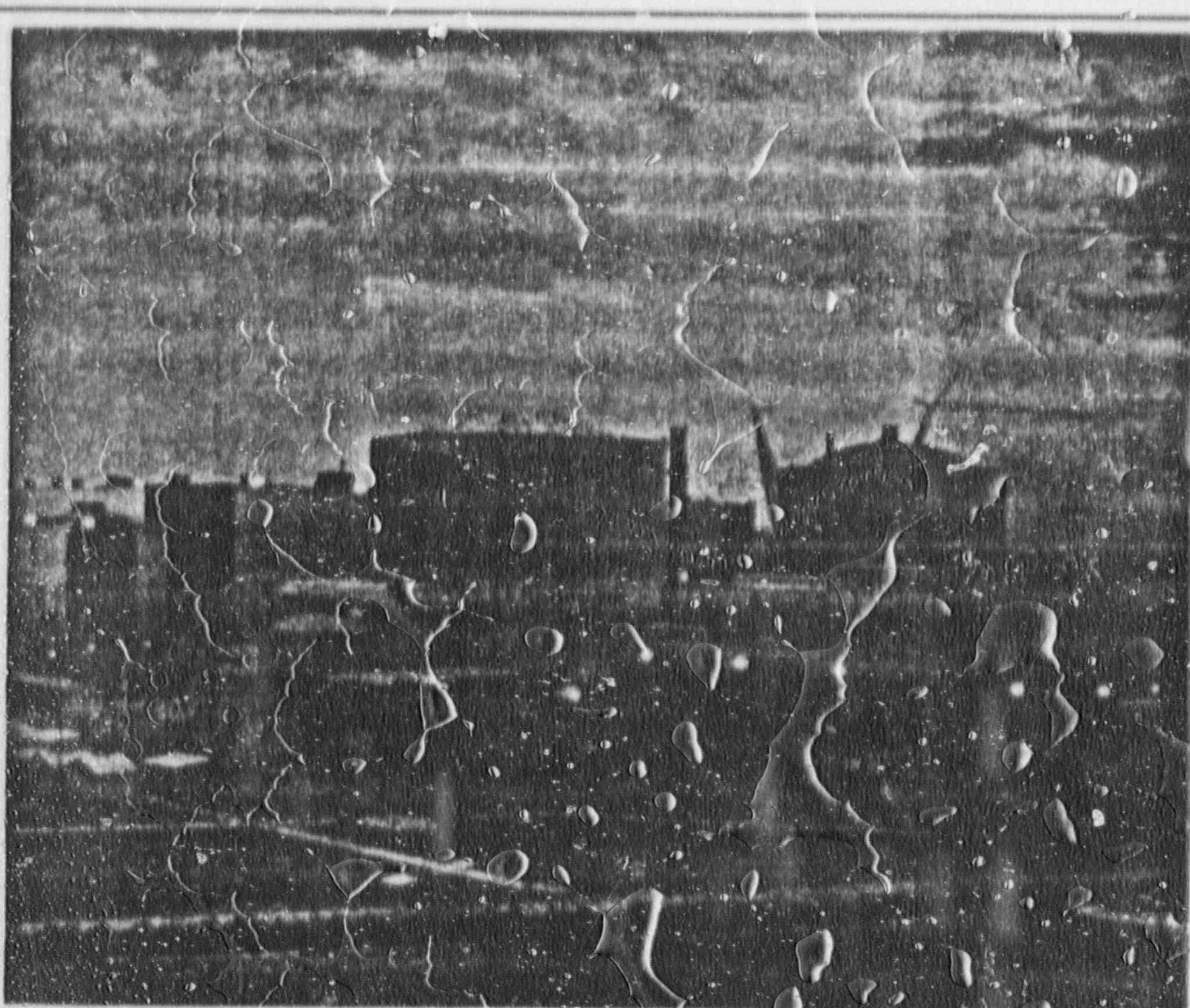
B&W FUEL COMPANY

BAW-2119

October 1990

RELOAD REPORT

Catawba Unit 1, Cycle 6



DUKE POWER COMPANY

Catawba Nuclear Station

B&W FUEL COMPANY