TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
1. Manual Reactor Trip	Not Applicable	Not Applicable	
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER	Low Setpoint - ≤ 27.4% of RATED THERMAL POWER	R145
	High Setpoint - ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 111.4% of RATED THERMAL POWER	R145
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second	≤ 6.3% of RATED THERMAL POWER with a time constant ≥ 2 second	
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second	≤ 6.3% of RATED THERMAL POWER with a time constant ≥ 2 second	
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 45.20% of RATED THERMAL POWER	R189
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.45 x 10^5 counts per second	R189
7. Overtemperature ΔT	See Note 1	See Note 3	
8. Overpower ΔT	See Note 2	See Note 4	
9. Pressurizer Pressure--Low	≥ 1970 psig	≥ 1964.8 psig	
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2390.2 psig	R145
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 92.7% of instrument span	
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89.6% of design flow per loop*	R225

*Design flow is 90,045 (87,000 X 1.035) gpm per loop.

NOTATION (Continued)

NOTE 1: (Continued)

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation R145

τ_1, & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, $\tau_1 \geq 33$ secs., $\tau_2 \leq 4$ secs. R215

T = Average temperature °F

T' ≤ 578.2°F (Nominal T_{avg} at RATED THERMAL POWER) R145

K_3 = 0.00055

P = Pressurizer pressure, psig

P' = 2235 psig (Nominal RCS operating pressure)

S = Laplace transform operator (sec^{-1})

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear 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 QTNL* and QTPL* $f_1(\Delta I) = 0$ (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).

NOTATION (Continued)

NOTE 1: (Continued)

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds QTNL*, the ΔT trip setpoint shall be automatically reduced by QTNS* of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds QTPL*, the ΔT trip setpoint shall be automatically reduced by QTPS* of its value at RATED THERMAL POWER.

NOTE 2: Overpower $\Delta T \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_3 S}{1 + \tau_3 S}\right) T - K_6 (T - T") - f_2(\Delta I)\}$ R145

Where: $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = as defined in Note 1

τ_4, τ_5 = as defined in Note 1

ΔT_o = as defined in Note 1

K_4 $\leq$ 1.087 R118

K_5 $\geq$ 0.02/°F for increasing average temperature and 0 for decreasing average temperature R215

$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation R145

* QTNL, QTPL, QTNS, and QTPS are specified in the COLR per Specification 6.9.1.14.

NOTE 2: (Continued)

τ_3 = Time constant utilized in the rate-lag controller for T_{avg}, $\tau_3 \geq 10$ secs. R215

K_6 $\geq$ 0.0011 for T > T" and $K_6 \geq 0$ for T $\leq$ T"

T = as defined in Note 1

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 578.2°F)

S = as defined in Note 1

and $f_2(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear 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 QPNL* and QPPL* $f_2(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds QPNL* the ΔT trip setpoint shall be automatically reduced by QPNS* of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds QPPL* the ΔT trip setpoint shall be automatically reduced by QPPS* of its value at RATED THERMAL POWER.

NOTE 3: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 1.9 percent ΔT span.

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 1.7 percent ΔT span. R145

*QPNL, QPPL, QPNS, and QPPS are specified in the COLR per Specification 6.9.1.14.

2.1.1 REACTOR CORE

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

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

The DNB design basis is that there must be at least a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the design DNBR limit. BR-4

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 percent probability at a 95 percent confidence level 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 uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analysis using values of input parameters without uncertainties.

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

The curves of Figure 2.1-1 are based on an enthalpy rise hot channel factor, $F^N_{\Delta H}$, (nominal values have been reduced to include a 4% total rod power uncertainty factor), 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:

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}[1+.3\ (1-P)]$$

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

$F^{RTP}_{\Delta H}$ = Nominal Values | 1.70 - Mark-BW Fuel
| 1.62 - Westinghouse Fuel |

Range Channels will initiate a reactor trip at approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature Delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to transit, thermowell, and RTD response time delays from the core to the temperature detectors (about 8 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1. R145 R145

The $f_1(\Delta I)$ trip reset term in the Overtemperature Delta T trip function precludes power distributions that cause the DNB limit to be exceeded during a limiting Condition II event. The negative and positive ΔI limits at which the $f_1(\Delta I)$ term begins to reduce the trip setpoint and the dependence of $f_1(\Delta I)$ on THERMAL POWER are determined on a cycle-specific basis using approved methodology and are specified in the COLR per Specification 6.9.1.14.

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature Delta T setpoint.

Delta-T_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured by the plant for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flows greater than thermal design flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Accurate determination of the loop specific ΔT value should be made quarterly and under steady state conditions (i.e., power distributions not affected by Xenon or other transient conditions). R145 BR-11

Overpower Delta T

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in axial power distribution,

density, and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. The setpoint is automatically reduced according to the notations in Table 2.2-1 to account for adverse axial flux differences.

The $f_2(\Delta I)$ trip reset term in the Overpower Delta T trip function precludes power distributions that cause the fuel melt limit to be exceeded during a limiting Condition II event. The negative and positive ΔI limits at which the $f_2(\Delta I)$ term begins to reduce the trip setpoint and the dependence of $f_2(\Delta I)$ on THERMAL POWER are determined on a cycle-specific basis using approved methodology and are specified in the COLR per Specification 6.9.1.14.

The Overpower Delta-T trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Delta-T_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured by the plant for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flows greater than thermal design flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Accurate determination of the loop specific ΔT value should be made quarterly and under steady state conditons (i.e., power distributions not affected by Xenon or other transient conditions). R145 BR-11

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drops below 90% of nominal full loop flow. Above the P-8 interlock, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the safety analysis DNBR limit during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature Delta T trip set point is adjusted to the value specified for all loops in operation. R145 R142 R145

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(X,Y,Z)$ shall be maintained within the acceptable limits specified in the COLR:

APPLICABILITY: MODE 1

ACTION:

With $F_Q(X,Y,Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(X,Y,Z)$ exceeds the limit within 15 minutes, and similarly reduce the following:

   1. Administratively reduce the allowable power at each point along the AFD limit lines within 2 hours, and

   2. The Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

b. POWER OPERATION may proceed for up to 48 hours. Subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% that $F_Q(X,Y,Z)$ exceeds the limit specified in the COLR.

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

R144

4.2.2.2 $F_Q^M(X,Y,Z)$ shall be evaluated to determine if $F_Q(X,Y,Z)$ is within its limit by:

a. Using the moveable incore detectors to obtain a power distribution map ($F_Q^M(X,Y,Z)$ *) at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Satisfying the following relationship:

$$F_Q^M(X,Y,Z) \leq BQNOM(X,Y,Z)$$

where BQNOM(X,Y,Z)** represents the nominal design increased by an allowance for the expected deviation between the nominal design and the measurement.

The BQNOM(X,Y,Z) factors 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.

c. If the above relationship is not satisfied, then

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

$$\%\ AFD\ Margin = \left(1 - \frac{F_Q^M(X,Y,Z)}{BQDES(X,Y,Z)}\right) x\ 100\%$$

$$\%\ f_2(\Delta I)\ Margin = \left(1 - \frac{F_Q^M(X,Y,Z)}{BCDES(X,Y,Z)}\right) x\ 100\%$$

where BQDES(X,Y,Z)** and BCDES(X,Y,Z)** represent the maximum allowable design peaking factors which insure that the licensing criteria will be preserved for operation within Limiting Condition for Operation limits, and include allowances for the calculational and measurement uncertainties.

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

** BQNOM (X,Y,Z), BQDES(X,Y,Z), and BCDES(X,Y,Z) Data bases are provided for input to the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in Specification 6.9.1.14.

SURVEILLANCE REQUIREMENTS (Continued)

2. Find the minimum margin of all locations examined in 4.2.2.2.c.1 above.

   AFD min margin = minimum % margin value of all locations examined.

   $f_2(\Delta I)$ OPΔT min margin = minimum % margin value of all locations examined.

3. If the AFD min margin in 4.2.2.2.c.2 above is <0, either the following actions shall be taken, or the action statements for 3.2.2 shall be followed.

   (a) Within 2 hours, administratively reduce the negative AFD limit lines at each power level by:

   Reduced AFD^{Limit} = (AFD^{Limit} from COLR) + absolute value of ($NSLOPE^{AFD}$* % x AFD min margin of 4.2.2.2.c.2)

   (b) Within 2 hours, administratively reduce the positive AFD limit lines at each power level by:

   Reduced AFD^{Limit} = (AFD^{Limit} from COLR) - absolute value of ($PSLOPE^{AFD}$* % X AFD min margin)

4. If the $f_2(\Delta I)$ min margin in 4.2.2.2.c.2 above is <0, either the following actions shall be taken, or the action statements for 3.2.2 shall be followed.

   (a) Within 48 hours, reduce the OPΔT negative $f_2(\Delta I)$ breakpoint limit by:

   Reduced OPΔT negative $f_2(\Delta I)$ breakpoint limit = ($f_2(\Delta I)$ limit of Table 2.2-1) + absolute value of

   $(NSLOPE^{f_2(\Delta I)}$** % x $f_2(\Delta I)$ min margin)

---

* $NSLOPE^{AFD}$ and $PSLOPE^{AFD}$ are the amount of AFD adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

** $NSLOPE^{f_2(\Delta I)}$ and $PSLOPE^{f_2(\Delta I)}$ are the amounts of the OPΔT $f_2(\Delta I)$ limit adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

(b) Within 48 hours, reduce the OPΔT positive $f_2(\Delta I)$ breakpoint limit by:

Reduced OPΔT positive $f_2(\Delta I)$ breakpoint limit = ($f_2(\Delta I)$ limit of Table 2.2-1) - absolute value of ($PSLOPE^{f_2(\Delta I)}$** % x f_2(Δ., min *margin*)

d. Measuring $F_Q^M(X,Y,Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(X,Y,Z)$ was last determined,*** or

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

e. With two measurements extrapolated to 31 EFPD beyond the most recent measurement yielding $F_Q^M(X,Y,Z) > BQNOM(X,Y,Z)$, either of the following actions specified shall be taken.

1. $F_Q^M(X,Y,Z)$ shall be increased over that specified in 4.2.2.2.a by the appropriate factor specified in the COLR, and 4.2.2.2.c repeated, or

2. $F_Q^M(X,Y,Z)$ shall be evaluated according to 4.2.2.2 at or before the time when the margin is projected to result in one of the actions specified in 4.2.2.2.c.3 or 4.2.2.2.c.4.

4.2.2.3 When $F_Q(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(X,Y,Z)$ shall be obtained from a power distribution map, increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty, and compared to the $F_Q(X,Y,Z)$ limit specified in the COLR according to Specification 3.2.2.

** $NSLOPE^{f_2(\Delta I)}$ and $PSLOPE^{f_2(\Delta I)}$ are the amounts of the OPΔT $f_2(\Delta I)$ limit adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

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

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}(X,Y)$ shall be maintained within the limits specified in the COLR

APPLICABILITY: MODE 1

ACTION:

With $F_{\Delta H}(X,Y)$ exceeding the limit specified in the COLR:

a. Within [illegible] hours either:

1. Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or

2. Reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH*% for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit, and

b. Within the next 4 hours either:

1. Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or

2. Reduce the Power Range Neutron Flux-High Trip Setpoint in Table 2.2-1 at least RRH*% for each 1% that $F_{\Delta H}(X,Y)$ exceeds that limit, and

c. Within 24 hours of initially being outside the limit specified in the COLR, either:

1. Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or

2. Verify through incore flux mapping that $F_{\Delta H}(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a.2 or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

---

* RRH is the amount of power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

ACTION: (Continued)

d. Within 48 hours of initially being outside the limit specified in the COLR, reduce the Overtemperature Delta T K_1 term in Table 2.2-1 by at least TRH** for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit, and

e. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2 and/or b. and/or c. and/or d., above: subsequent POWER OPERATION may proceed provided that $F_{\Delta H}(X,Y)$ is demonstrated, through incore flux mapping, to be within the above limit prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

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

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

** TRH is the amount of Overtemperature Delta T K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

SURVEILLANCE REQUIREMENTS

4.?.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^M_{\Delta H}(X,Y)$ shall be evaluated to determine if $F_{\Delta H}(X,Y)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map $F^M_{\Delta H}(X,Y)$ * at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Satisfying the following relationship:

$F\Delta HR^M(X,Y) \leq BHNOM(X,Y)$

Where:

$$F\Delta HR^M(X,Y) = \frac{F^M_{\Delta H}(X,Y)}{MAP^M / AXIAL(X,Y)}$$

And BHNOM(X,Y)** represents the nominal design increased by an allowance for the expected deviation between the nominal design and the measurement.

MAP^M is the maximum Allowable Peak** obtained from the measured power distribution.

AXIAL(X,Y) is the axial shape for $F_{\Delta H}(X,Y)$.

c. If the above relationship is not satisfied, then

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

$$\%\ F_{\Delta H}\ Margin = \left(1 - \frac{F\Delta HR^M(X,Y)}{BHDES(X,Y)}\right) x\ 100\%$$

$$\%\ f_1(\Delta I)\ Margin = \left(1 - \frac{F\Delta HR^M(X,Y)}{BRDES(X,Y)}\right) \times 100\%$$

where BHDES(X,Y) and BRDES(X,Y)** represent the maximum allowable design peaking factors which insure that the licensing criteria will be preserved for operation within the LCO limits, and include allowances for calculational and measurement uncertainties.

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* No additional uncertainties are required in the following equations for $F^M_{\Delta H}(X,Y)$ and $F\Delta HR^M(X,Y)$, because the limits include uncertainties.

** BHNOM(X,Y), MAP^M, BHDES(X,Y), and BRDES(X,Y) data bases are provided for input to the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in Specification 6.9.1.14.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^M_{\Delta H}(X,Y)$ shall be evaluated to determine if $F_{\Delta H}(X,Y)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map $F^M_{\Delta H}(X,Y)$* at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Satisfying the following relationship:

$F\Delta HR^M(X,Y) \leq BHNOM(X,Y)$

Where:

$$F\Delta HR^M(X,Y) = \frac{F^M_{\Delta H}(X,Y)}{MAP^M / AXIAL(X,Y)}$$

And BHNOM(X,Y)** represents the nominal design increased by an allowance for the expected deviation between the nominal design and the measurement.

MAP^M is the maximum Allowable Peak** obtained from the measured power distribution.

AXIAL(X,Y) is the axial shape for $F_{\Delta H}(X,Y)$.

c. If the above relationship is not satisfied, then

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

$$\%\ F_{\Delta H}\ Margin = \left(1 - \frac{F\Delta HR^M(X,Y)}{BHDES(X,Y)}\right) x\ 100\%$$

$$\%\ f_1(\Delta I)\ Margin = \left(1 - \frac{F\Delta HR^M(X,Y)}{BRDES(X,Y)}\right) \times 100\%$$

where BHDES(X,Y) and BRDES(X,Y)** represent the maximum allowable design peaking factors which insure that the licensing criteria will be preserved for operation within the LCO limits, and include allowances for calculational and measurement uncertainties.

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* No additional uncertainties are required in the following equations for $F^M_{\Delta H}(X,Y)$ and $F\Delta HR^M(X,Y)$, because the limits include uncertainties.

** BHNOM(X,Y), MAP^M, BHDES(X,Y), and BRDES(X,Y) data bases are provided for input to the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in Specification 6.9.1.14.

SURVEILLANCE REQUIREMENTS

2. Find the minimum margin of all locations examined in 4.2.3.2.c.1 above.

   $F_{\Delta H}$ min margin = minimum % margin value of all locations examined

   $f_1(\Delta I)$ min margin = minimum % margin value of all locations examined

3. If the $F_{\Delta H}$ min margin in 4.2.3.2.c.2 above is < 0, then within 2 hours reduce the allowable THERMAL POWER from RATED THERMAL POWER by RRH*% x most negative margin from 4.2.3.2.c.2 and maintain the requirements of Specification 3.2.3; otherwise the Action statements for 3.2.3 apply.

4. If the $f_1(\Delta I)$ min margin in 4.2.3.2.c.2 above is < 0, then within 48 hours reduce the Overtemperature Delta T K1 term in Table 2.2-1 by at least TRH**% x most negative margin from 4.2.3.2.c.2 and maintain the requirements of Specification 3.2.3; otherwise the action statements for 3.2.3 apply.

d. With two measurements extrapolated to 31 EFPD beyond the most recent measurement yielding

   $F\Delta HR^M(X,Y) > BHNOM(X,Y)$

   either of the following actions shall be taken:

   1. $F^M_{\Delta H}(X,Y)$ shall be increased over that specified in 4.2.3.2.a by the appropriate factor specified in the COLR, and 4.2.3.2.c.1 repeated, or

   2. $F^M_{\Delta H}(X,Y)$ shall be evaluated according to 4.2.3.2 at or before the time when the margin is projected to result in the action specified in 4.2.3.2.c.3 or 4.2.3.2.c.4.

4.2.3.3 $F^M_{\Delta H}(X,Y)$ shall be determined to be within its limit by using the incore detectors to obtain a power distribution map:

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

b. At least once per 31 EFPD.

---

* RRH is the amount of power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

** TRH is the amount of Overtemperature Delta T K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta h}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

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

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

      a) Either 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:

      a) Either 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 1.02 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.

   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 Exception 3.10.2.

ACTION: (Continued)

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

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

a) Either 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 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.

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:

a) Either 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.

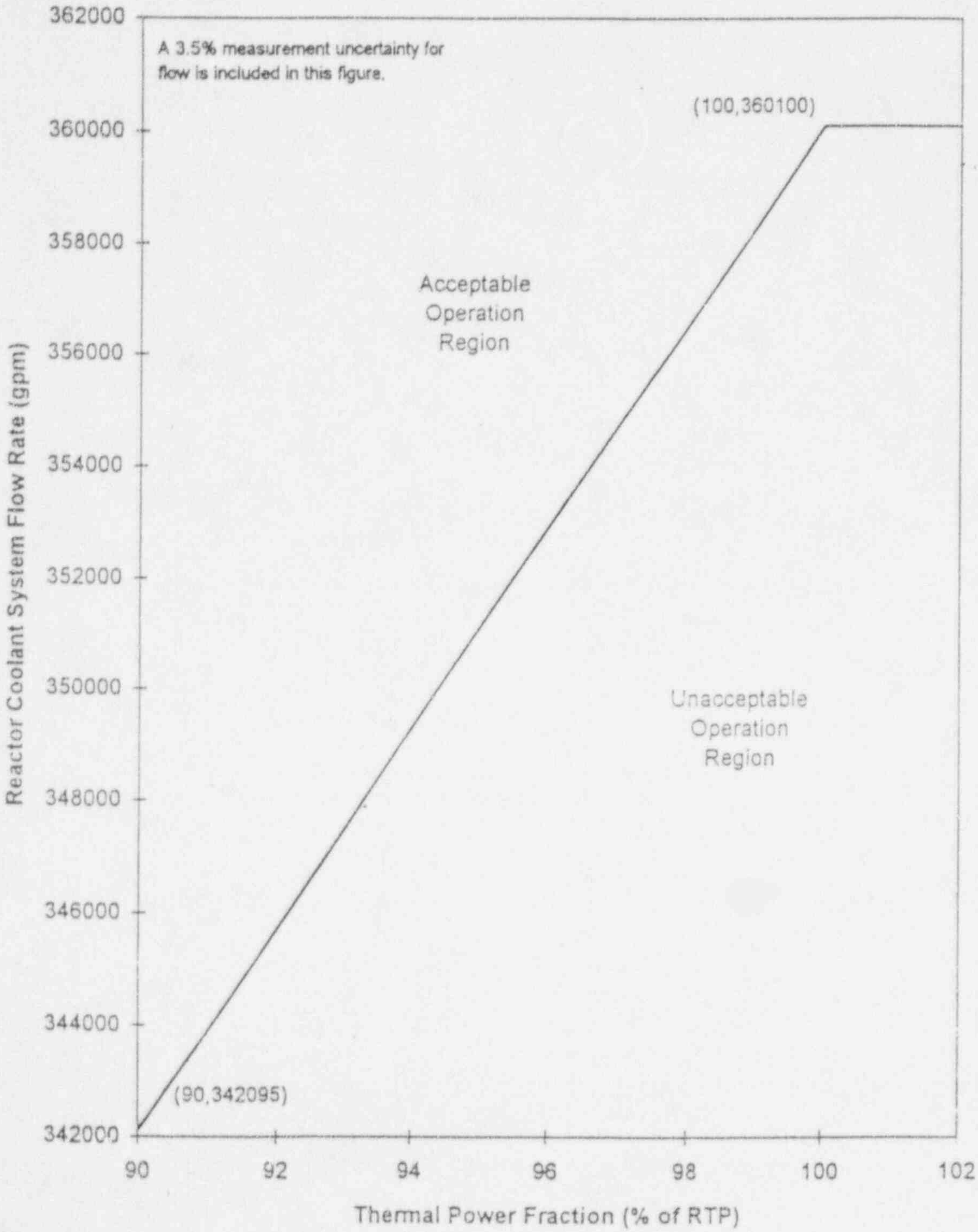
TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	≤ 583°F
Pressurizer Pressure	≥ 2220 psia*
Reactor Coolant System Total Flow	Figure 3.2-1

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

Figure 3.2-1 Flow vs. Power for 4 Loops in Operation



The control rod insertion limits and shutdown rod insertion limits are specified in the COLR per Specification 6.9.1.14.

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

In the event that a malfunction of the Rod Control System renders control rods immovable, provision is made for continued operation provided: R219

- o The affected control rods remain trippable, and
- o The individual control rod alignment limits are met.

In the event that a malfunction of the Rod Control System renders control rod banks immovable during surveillance testing, provision is made for 72 hours of continued operation provided:

- o The affected control rod banks remains trippable,
- o The individual control rod alignment limits are met,
- o A maximum of one control or shutdown bank is inserted no more than 18 steps below the insertion limit,
- o No reactor coolant system boron concentration dilution activities or power level increases are allowed, and
- o The SHUTDOWN MARGIN requirements are verified every 12 hours or upon insertion of controlling bank during the period the insertion limit is not met.

The requirements to preclude Reactor Coolant System boron concentration dilution, while a control or shutdown bank is below insert limits, will minimize the impact on shutdown margin.

The controlling bank(s), which is normally Control Bank D, is excluded from the 72-hour provision since insertion of this bank(s) below the insertion limit is not required for control rod assembly surveillance testing. A controlling bank is defined as any control bank that is less than fully withdrawn as defined in the COLR with the exception of fully withdrawn banks that have been inserted in accordance with Surveillance Requirement 4.1.3.1.2. This provision excludes the use of the 72-hour allowance for control banks that can be exercised 10 steps in either direction without exceeding the insertion limits.

Checks are performed for each reload core to ensure that bank insertions of up to 18 steps will not result in power distributions, which violate the DNB criterion for ANS Condition II transients (moderate frequency transients analyzed in Section 15.2 of the UFSAR). Administrative requirements on the initial controlling bank position will ensure that this insertion and an additional controlling bank insertion of five steps or less will not violate the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 during the repair period. If the controlling bank is inserted more than five steps deeper than its initial position, a calculation will be performed to ensure that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is met. Since no dilution or power level increases are allowed, shutdown margin will be maintained as long as the controlling bank is far enough above its insertion limit to compensate for the inserted worth of the bank that is beyond its insertion limit.

The 72-hour period for a control rod assembly bank to be inserted below its insertion limit restricts the likelihood of a more severe (i.e., ANS Condition III or IV) accident or transient condition occurring concurrently with the insertion limit violation.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) 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:

$F_Q(X,Y,Z)$ 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.

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

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(X,Y,Z)$ upper bound envelope of F_Q limit specified in the COLR times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes. R15 R144

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. 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 outside the allowed ΔI-Power operating space and the THERMAL POWER is greater than 50 percent of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTORS

The limits on the heat flux hot channel factor and the 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. The peaking limits are specified in the COLR per Specification 6.9.1.14. R142

Each of these hot channel factors 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 ± 13 steps from the group demand position.

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

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The $F_{\Delta H}(X,Y)$ limit as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F_{\Delta H}(X,Y)$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an $F_Q(X,Y,Z)$ measurement is taken, an allowance for measurement uncertainty is made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and this allowance is included in the methodology applied to the determination of the core operating limits as described in the reference in Specification 6.9.1.14.

The hot channel factors, $F_Q^M(X,Y,Z)$ and $F_{\Delta H}^M(X,Y)$, are measured periodically and compared to the nominal design values to provide a reasonable assurance that the core is operating as designed and that the limiting criteria will not be exceeded for operation within the Technical Specification limits of Sections 2.2 (Limiting Safety System Settings), 3.1.3 (Moveable Control Assemblies), 3.2.1 (AXIAL FLUX DIFFERENCE), and 3.2.4 (QUADRANT POWER TILT RATIO). An allowance is provided to account for the expected deviation between the calculation and the measurement. If the measurement is above the maximum expected value for that location, it is assumed to not be operating as designed, and a peaking margin evaluation is performed to provide a basis for decreasing the width of the AFD and f(ΔI) limits, and for reducing THERMAL POWER.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that no anomaly exists such 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 QUADRANT POWER TILT RATIO limit at which corrective action is required provides DNB and linear heat generation protection with x-y plane power tilts. The QUADRANT POWER TILT RATIO limit is reflected by a corresponding peaking augmentation factor which is included in the generation of the AFD limits.

The 2-hour time allowance for operation with the 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 $F_Q(X,Y,Z)$ is reinstated by reducing the allowable THERMAL POWER by 3 percent for each percent of tilt in excess of 1.02.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the safety analysis DNBR limit throughout each analyzed transient. R142

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.

The flow parameters indicated in Figure 3.2-1 have been rounded down to bias the analysis in the conservative direction.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable in the updated final safety analysis report. R194

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

Action 15 of Table 3.3-1, Reactor Trip System Instrumentation, allows the breaker to be bypassed for up to 4 hours for the purpose of performing maintenance. The 4 hours is based on a Westinghouse analysis performed in WCAP-10271, Supplement 1, which determines bypass breaker availability. R58

The placing of a channel in the trip condition provides the safety function of the channel. If the channel is tripped for testing and no other condition would have indicated inoperability, the channel should not be declared inoperable. BR-9

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(X,Y,Z)$ or $F_{\Delta H}(X,Y)$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 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 no later than the 15th of each month following the calendar month covered by the report. R76

CORE OPERATING LIMITS REPORT R159

6.9.1.14 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. $f_1(\Delta I)$ limits for Overtemperature Delta T Trip Setpoints and $f_2(\Delta I)$ limits for Overpower Delta T Trip Setpoints for Specification 2.2.1.

2. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

3. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

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

5. AXIAL FLUX DIFFERENCE Limits for Specification 3/4.2.1,

6. Heat Flux Hot Channel Factor and K(z) for Specification 3/4.2.2, and

7. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3.

6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in: R159

1. BAW-10180P-A, Rev. 1, "NEMO - NODAL EXPANSION METHOD OPTIMIZED", March 1993. (FCF Proprietary)
   (Methodology for Specification 3.1.1.3-Moderator Temperature Coefficient)

2. BAW-10169P-A, "RSG PLANT SAFETY ANALYSIS - B&W SAFETY ANALYSIS METHODOLOGY FOR RECIRCULATING STEAM GENERATOR PLANTS", October 1989. (FCF Proprietary)
   (Methodology for Specification 3.1.1.3-Moderator Temperature Coefficient)

3. BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June 1989. (FCF Proprietary)
   (Methodology for Specification 2.2.1, - Limiting Safety System Settings [$f_1(\Delta I)$, $f_2(\Delta I)$ limits], 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3/4.2.1 - Axial Flux Difference, 3/4.2.2 - Heat Flux Hot Channel Factor, 3/4.2.3 - Nuclear Enthalpy Rise Hot Channel Factor)

4. BAW-10168P-A, Rev. 2, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

5. BAW-10168P-A, Rev 3, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

CORE OPERATING LIMITS REPORT (continued)

6. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985, (W Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)
7. WCAP-10266-P-A, Rev. 2, "THE 1981 REVISION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

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

6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days after cycle start-up (Mode 2) for each reload cycle or within 30 days of issuance of any midcycle revision of the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. R159

SPECIAL REPORTS

6.9.2.1 Special reports shall be submitted within the time period specified for each report, in accordance with 10 CFR 50.4. R76

6.9.2.2 Diesel Generator Reliability Improvement Program

As a minimum the Reliability Improvement Program report for NRC audit, required by LCO 3.8.1.1, Table 4.8-1, shall include:

(a) a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed R56

(b) analysis of failures and determination of root causes of failures

(c) evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the Plant

(d) identification of all actions taken or to be taken to 1) correct the root causes of failures defined in b) above and 2) achieve a general improvement of diesel generator reliability

(e) the schedule for implementation of each action from d) above

(f) an assessment of the existing reliability of electric power to engineered-safety-feature equipment

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
1. Manual Reactor Trip	Not Applicable	Not Applicable	
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER	Low Setpoint - ≤ 27.4% of RATED THERMAL POWER	R132
	High Setpoint - ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 111.4% of RATED THERMAL POWER	R132
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 6.3% of RATED THERMAL POWER with a time constant ≥ 2 seconds	R36
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 6.3% of RATED THERMAL POWER with a time constant ≥ 2 seconds	R36
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 45.20% of RATED THERMAL POWER	R177
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.45 x 10^5 counts per second	R177
7. Overtemperature ΔT	See Note 1	See Note 3	
8. Overpower ΔT	See Note 2	See Note 4	R132
9. Pressurizer Pressure--Low	≥ 1970 psig	≥ 1964.8 psig	
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2390.2 psig	R203
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 92.7% of instrument span	
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89.6% of design flow per loop*	R212

*Design flow is 90,045 (87,000 x 1.035) gpm per loop.

NOTATION (Continued)

NOTE 1: (Continued)

S = Laplace transform operator, $\sec^{-1}$

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear 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 QTNL* and QTPL* $f_1(\Delta I) = 0$ (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). R21

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds QTNL*, the ΔT trip setpoint shall be automatically reduced by QTNS* of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds QTPL*, the ΔT trip setpoint shall be automatically reduced by QTPS* of its value at RATED THERMAL POWER.

NOTE 2: Overpower $\Delta T \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right) \le \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_3 S}{1 + \tau_3 S}\right) T - K_6 [T - T''] - f_2(\Delta I)\right\}$

where: $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = as defined in Note 1 R132

*QTNL, QTPL, QTNS, and QTPS are specified in the COLR per Specification 6.9.1.14.

NOTATION (Continued)

NOTE 2: (Continued)

τ_4, τ_5 = as defined in Note 1 | R132

ΔT_o = as defined in Note 1

K_4 ≤ 1.087 | R104

K_5 ≥ 0.02/°F for increasing average temperature and 0 for decreasing average temperature | R201

$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation | R132

τ_3 = Time constant utilized in the rate-lag controller for T_{avg}, $\tau_3 \geq 10$ secs. | R201

K_6 ≥ 0.0011 for T > T" and $K_6 \geq 0$ for T ≤ T"

T = as defined in Note 1

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 578.2°F)

S = as defined in Note 1

and $f_2(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear 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 QPNL* and QPPL* $f_2(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds QPNL* the ΔT trip setpoint shall be automatically reduced by QPNS* of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds QPPL* the ΔT trip setpoint shall be automatically reduced by QPPS* of its value at RATED THERMAL POWER.

*QPNL, OPPL, QPNS, and OPPS are specified in the COLR per Specification 6.9.1.14.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and; therefore, THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is that there must be at least a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the design DNBR limit. BR-5

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 percent probability at a 95 percent confidence level 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 uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analysis using values of input parameters without uncertainties.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. R104 R130

The curves of Figure 2.1-1 are based on an enthalpy rise hot channel factor, $F^N_{\Delta H}$, (nominal values have been reduced to include a 4% total rod power uncertainty factor), 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: R21

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}[1+.3\ (1-P)]$$

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

R146

$F^{RTP}_{\Delta H}$ = Nominal Values

1.70	Mark-BW Fuel
1.62	Westinghouse Fuel

Intermediate and Source Range, Nuclear Flux (Continued)

Range Channels will initiate a reactor trip at approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System. R129

Overtemperature ΔT

The Overtemperature Delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to transit, thermowell, and RTD response time delays from the core to the temperature detectors (about 8 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature and dynamic compensation for transport, thermowell, and RTD response time delays from the core to the RTD output indication. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1. R132

The $f_1(\Delta I)$ trip reset term in the Overtemperature Delta T trip function precludes power distributions that cause the DNB limit to be exceeded during a limiting Condition II event. The negative and positive ΔI limits at which the $f_1(\Delta I)$ term begins to reduce the trip setpoint and the dependence of $f_1(\Delta I)$ on THERMAL POWER are determined on a cycle-specific basis using approved methodology and are specified in the COLR per Specification 6.9.1.14.

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature Delta T setpoint.

Delta-T_o, as used in the Overtemperature and Overpower ΔT trips, represents the 100 percent RTP value as measured by the plant for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flows greater than thermal design flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Accurate determination of the loop specific ΔT value should be made when performing Incore/Excore quarterly recalibration and under steady state conditions (i.e., power distributions not affected by xenon or other transient conditions.). R132

Overpower ΔT

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in axial power distribution, density and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to the RTD output indication. The setpoint is automatically reduced according to the notations in Table 2.2-1 to account for adverse axial flux differences.

The $f_2(\Delta I)$ trip reset term in the Overpower Delta T trip function precludes power distributions that cause the fuel melt limit to be exceeded during a limiting Condition II event. The negative and positive ΔI limits at which the $f_2(\Delta I)$ term begins to reduce the trip setpoint and the dependence of $f_2(\Delta I)$ on THERMAL POWER are determined on a cycle-specific basis using approved methodology and are specified in the COLR per Specification 6.9.1.14.

The Overpower Delta T trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Delta-T_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100 percent RTP value as measured by the plant for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flows greater than thermal design flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Accurate determination of the loop specific ΔT value R132
should be made when performing Incore/Excore quarterly recalibration and under steady state conditions (i.e., power distributions not affected by xenon or other transient conditions.).

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(X,Y,Z)$ shall be maintained within the acceptable limits specified in the COLR.

APPLICABILITY: MODE 1 R21

ACTION:

With $F_Q(X,Y,Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(X,Y,Z)$ exceeds the limit within 15 minutes, and similarly reduce the following:

   1. Administratively reduce the allowable power at each point along the AFD limit lines within 2 hours, and

   2. The Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

b. POWER OPERATION may proceed for up to 48 hours. Subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% that $F_Q(X,Y,Z)$ exceeds the limit specified in the COLR.

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

SURVEILLANCE REQUIREMENTS R21

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q^M(X,Y,Z)$ shall be evaluated to determine if $F_Q(X,Y,Z)$ is within its limit by:

a. Using the moveable incore detectors to obtain a power distribution map ($F_Q^M(X,Y,Z)$ *) at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Satisfying the following relationship:

$$F_Q^M(X,Y,Z) \leq BQNOM(X,Y,Z)$$

where BQNOM(X,Y,Z)** represents the nominal design increased by an allowance for the expected deviation between the nominal design and the measurement.

The BQNOM(X,Y,Z) factors 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.

c. If the above relationship is not satisfied, then

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

$$\%\ AFD\ Margin = \left(1 - \frac{F_Q^M(X,Y,Z)}{BQDES(X,Y,Z)}\right) x\ 100\%$$

$$\%\ f_2(\Delta I)\ Margin = \left(1 - \frac{F_Q^M(X,Y,Z)}{BCDES(X,Y,Z)}\right) x\ 100\%$$

where BQDES(X,Y,Z)** and BCDES(X,Y,Z)** represent the maximum allowable design peaking factors which insure that the licensing criteria will be preserved for operation within Limiting Condition for Operation limits, and include allowances for the calculational and measurement uncertainties.

---

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

** BQNOM (X,Y,Z), BQDES(X,Y,Z), and BCDES(X,Y,Z) Data bases are provided for input to the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in Specification 6.9.1.14.

SURVEILLANCE REQUIREMENTS (Continued)

2. Find the minimum margin of all locations examined in 4.2.2.2.c.1 above.

   AFD min margin = minimum % margin value of all locations examined.

   $f_2(\Delta I)$ OPΔT min margin = minimum % margin value of all locations examined.

3. If the AFD min margin in 4.2.2.2.c.2 above is <0, either the following actions shall be taken, or the action statements for 3.2.2 shall be followed.

   (a) Within 2 hours, administratively reduce the negative AFD limit lines at each power level by:

   Reduced AFD^{Limit} = (AFD^{Limit} from COLR) + absolute value of ($NSLOPE^{AFD}$* % x AFD min margin of 4.2.2.2.c.2)

   (b) Within 2 hours, administratively reduce the positive AFD limit lines at each power level by:

   Reduced AFD^{Limit} = (AFD^{Limit} from COLR) - absolute value of ($PSLOPE^{AFD}$* % X AFD min margin)

4. If the $f_2(\Delta I)$ min margin in 4.2.2.2.c.2 above is <0, either the following actions shall be taken, or the action statements for 3.2.2 shall be followed.

   (a) Within 48 hours, reduce the OPΔT negative $f_2(\Delta I)$ breakpoint limit by:

   Reduced OPΔT negative $f_2(\Delta I)$ breakpoint limit = ($f_2(\Delta I)$ limit of Table 2.2-1) + absolute value of

   $$(NSLOPE^{f_2(\Delta I)**}\ \%\ x\ f_2(\Delta I)\ \text{min margin})$$

---

* $NSLOPE^{AFD}$ and $PSLOPE^{AFD}$ are the amount of AFD adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

** $NSLOPE^{f_2(\Delta I)}$ and $PSLOPE^{f_2(\Delta I)}$ are the amounts of the OPΔT $f_2(\Delta I)$ limit adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

(b) Within 48 hours, reduce the OPΔT positive $f_2(\Delta I)$ breakpoint limit by:

Reduced OPΔT positive $f_2(\Delta I)$ breakpoint limit = ($f_2(\Delta I)$ limit of Table 2.2-1) - absolute value of ($PSLOPE^{f_2(\Delta I)}$** % x $f_2(\Delta I)$ min *margin*)

d. Measuring $F_Q^M(X,Y,Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(X,Y,Z)$ was last determined,*** or

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

e. With two measurements extrapolated to 31 EFPD beyond the most recent measurement yielding $F_Q^M(X,Y,Z) > BQNOM(X,Y,Z)$, either of the following actions specified shall be taken.

1. $F_Q^M(X,Y,Z)$ shall be increased over that specified in 4.2.2.2.a by the appropriate factor specified in the COLR, and 4.2.2.2.c repeated, or

2. $F_Q^M(X,Y,Z)$ shall be evaluated according to 4.2.2.2 at or before the time when the margin is projected to result in one of the actions specified in 4.2.2.2.c.3 or 4.2.2.2.c.4.

4.2.2.3 When $F_Q(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(X,Y,Z)$ shall be obtained from a power distribution map, increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty, and compared to the $F_Q(X,Y,Z)$ limit specified in the COLR according to Specification 3.2.2.

** $NSLOPE^{f_2(\Delta I)}$ and $PSLOPE^{f_2(\Delta I)}$ are the amounts of the OPΔT $f_2(\Delta I)$ limit adjustment required to compensate for each 1% that $F_Q(X,Y,Z)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

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

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}(X,Y)$ shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1

ACTION:

With $F_{\Delta H}(X,Y)$ exceeding the limit specified in the COLR:

a. Within 2 hours either:

1. Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or
2. Reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH*% for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit, and

b. Within the next 4 hours either:

1. Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or
2. Reduce the Power Range Neutron Flux-High Trip Setpoint in Table 2.2-1 at least RRH*% for each 1% that $F_{\Delta H}(X,Y)$ exceeds that limit, and

c. Within 24 hours of initially being outside the limit specified in the COLR, either:

1. Restore $F_{\Delta H}(X,Y)$ to within the limit specified in the COLR, or
2. Verify through incore flux mapping that $F_{\Delta H}(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a.2 or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

---

* RRH is the amount of power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

ACTION: (Continued)

d. Within 48 hours of initially being outside the limit specified in the COLR, reduce the Overtemperature Delta T K_1 term in Table 2.2-1 by at least TRH** for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit, and

e. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2 and/or b. and/or c. and/or d., above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}(X,Y)$ is demonstrated, through incore flux mapping, to be within the above limit prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

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

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

** TRH is the amount of Overtemperature Delta T K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^M_{\Delta H}(X,Y)$ shall be evaluated to determine if $F_{\Delta H}(X,Y)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map $F^M_{\Delta H}(X,Y)$* at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Satisfying the following relationship:

$F\Delta HR^M(X,Y) \le BHNOM(X,Y)$

Where:

$$F\Delta HR^M(X,Y) = \frac{F^M_{\Delta H}(X,Y)}{MAP^M / AXIAL(X,Y)}$$

And BHNOM(X,Y)** represents the nominal design increased by an allowance for the expected deviation between the nominal design and the measurement.

MAP^M is the maximum Allowable Peak** obtained from the measured power distribution.

AXIAL(X,Y) is the axial shape for $F_{\Delta H}(X,Y)$.

c. If the above relationship is not satisfied, then

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

$$\%\ F_{\Delta H}\ Margin = \left(1 - \frac{F\Delta HR^M(X,Y)}{BHDES(X,Y)}\right) x\ 100\%$$

$$\%\ f_1(\Delta I)\ Margin = \left(1 - \frac{F\Delta HR^M(X,Y)}{BRDES(X,Y)}\right) \times 100\%$$

where BHDES(X,Y) and BRDES(X,Y)** represent the maximum allowable design peaking factors which insure that the licensing criteria will be preserved for operation within the LCO limits, and include allowances for calculational and measurement uncertainties.

* No additional uncertainties are required in the following equations for $F^M_{\Delta H}(X,Y)$ and $F\Delta HR^M(X,Y)$, because the limits include uncertainties.

** BHNOM(X,Y), MAP^M, BHDES(X,Y), and BRDES(X,Y) data bases are provided for input to the plant power distribution analysis computer codes on a cycle specific basis and are determined using the methodology for core limit generation described in the references in Specification 6.9.1.14.

2. Find the minimum margin of all locations examined in 4.2.3.2.c.1 above.

   $F_{\Delta H}$ min margin = minimum % margin value of all locations examined

   $f_1(\Delta I)$ min margin = minimum % margin value of all locations examined

3. If the $F_{\Delta H}$ min margin in 4.2.3.2.c.2 above is < 0, then within 2 hours reduce the allowable THERMAL POWER from RATED THERMAL POWER by RRH*% x most negative margin from 4.2.3.2.c.2 and maintain the requirements of Specification 3.2.3; otherwise the Action statements for 3.2.3 apply.

4. If the $f_1(\Delta I)$ min margin in 4.2.3.2.c.2 above is < 0, then within 48 hours reduce the Overtemperature Delta T K1 term in Table 2.2-1 by at least TRH**% x most negative margin from 4.2.3.2.c.2 and maintain the requirements of Specification 3.2.3; otherwise the action statements for 3.2.3 apply.

d. With two measurements extrapolated to 31 EFPD beyond the most recent measurement yielding

   $F\Delta HR^{M}(X,Y) > BHNOM(X,Y)$

   either of the following actions shall be taken:

   1. $F^{M}_{\Delta H}(X,Y)$ shall be increased over that specified in 4.2.3.2.a by the appropriate factor specified in the COLR, and 4.2.3.2.c.1 repeated, or

   2. $F^{M}_{\Delta H}(X,Y)$ shall be evaluated according to 4.2.3.2 at or before the time when the margin is projected to result in the action specified in 4.2.3.2.c.3 or 4.2.3.2.c.4.

4.2.3.3 $F^{M}_{\Delta H}(X,Y)$ shall be determined to be within its limit by using the incore detectors to obtain a power distribution map:

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

b. At least once per 31 EFPD.

---

* RRH is the amount of power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

** TRH is the amount of Overtemperature Delta T K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds the limit provided in the COLR per Specification 6.9.1.14.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

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

   1. Calculate the QUARANT 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 1.02 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.

   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 Exception 3.10.2.

ACTION: (Continued)

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

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

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

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

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02 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.

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.

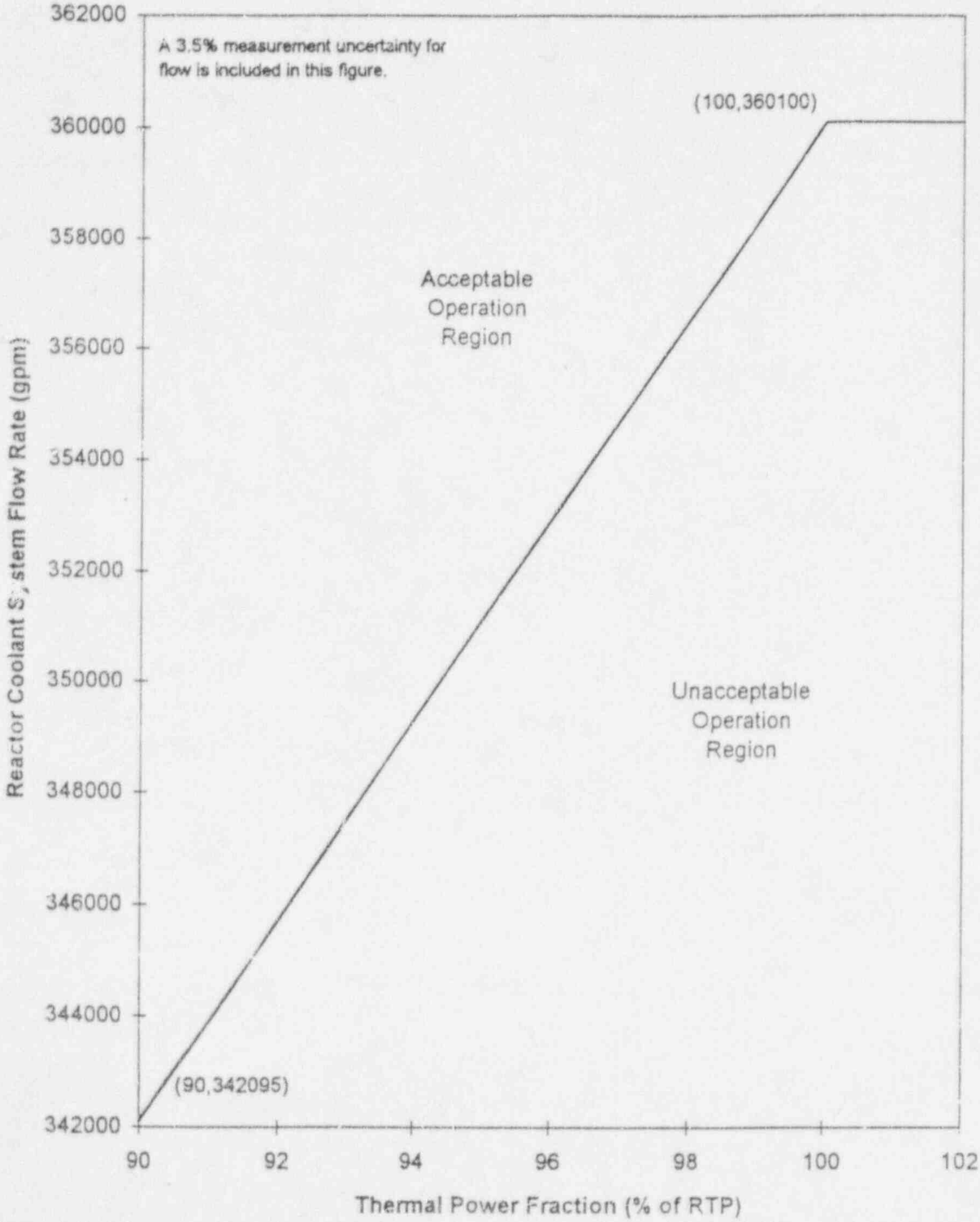
TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 583°F
Pressurizer Pressure	$\geq$ 2220 psia*
Reactor Coolant System 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, physics test, or performance of surveillance requirement 4.1.1.3.b.

Figure 3.2-1 Flow vs. Power for 4 Loops in Operation



MOVEABLE CONTROL ASSEMBLIES (Continued)

The control rod insertion limits and shutdown rod insertion limits are specified in the COLR for Specification 6.9.1.14.

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

In the event that a malfunction of the Rod Control System renders control rods immovable, provision is made for continued operation provided: R205

- o The affected control rods remain trippable, and
- o The individual control rod alignment limits are met.

In the event that a malfunction of the Rod Control System renders control rod banks immovable during surveillance testing, provision is made for 72 hours of continued operation provided:

- o The affected control rod banks remains trippable,
- o The individual control rod alignment limits are met,
- o A maximum of one control or shutdown bank is inserted no more than 18 steps below the insertion limit,
- o No reactor coolant system boron concentration dilution activities or power level increases are allowed, and
- o The SHUTDOWN MARGIN requirements are verified every 12 hours or upon insertion of controlling bank during the period the insertion limit is not met.

The requirements to preclude Reactor Coolant System boron concentration dilution, while a control or shutdown bank is below insert limits, will minimize the impact on shutdown margin.

The controlling bank(s), which is normally Control Bank D, is excluded from the 72-hour provision since insertion of this bank(s) below the insertion limit is not required for control rod assembly surveillance testing. A controlling bank is defined as any control bank that is less than fully withdrawn as defined in the COLR with the exception of fully withdrawn banks that have been inserted in accordance with Surveillance Requirement 4.1.3.1.2. This provision excludes the use of the 72-hour allowance for control banks that can be exercised 10 steps in either direction without exceeding the insertion limits.

Checks are performed for each reload core to ensure that bank insertions of up to 18 steps will not result in power distributions, which violate the DNB criterion for ANS Condition II transients (moderate frequency transients analyzed in Section 15.2 of the UFSAR). Administrative requirements on the initial controlling bank position will ensure that this insertion and an additional controlling bank insertion of five steps or less will not violate the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 during the repair period. If the controlling bank is inserted more than five steps deeper than its initial position, a calculation will be performed to ensure that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is met. Since no dilution or power level increases are allowed, shutdown margin will be maintained as long as the controlling bank is far enough above its insertion limit to compensate for the inserted worth of the bank that is beyond its insertion limit.

The 72-hour period for a control rod assembly bank to be inserted below its insertion limit restricts the likelihood of a more severe (i.e., ANS Condition III or IV) accident or transient condition occurring concurrently with the insertion limit violation.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) 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. R21

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

$F_Q(X,Y,Z)$ 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. R21

$F_{\Delta H}(X,Y)$ 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. R21

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(X,Y,Z)$ upper bound envelope of the F_Q limit specified in the COLR times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes. R146 R21

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The compuer 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 outside the allowed ΔI-Power operating space and the THERMAL POWER is greater than 50 percent of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTORS

The limits on the heat flux hot channel factor and the 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. The peaking limits are specified in the COLR per Specification 6.9.1.14. R21

Each of these hot channel factors 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: R130 R21

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 13 steps from the group demand position.

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

c. The control rod insertion limits of specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The $F_{\Delta H}(X,Y)$ limit as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F_{\Delta H}(X,Y)$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an $F_Q(X,Y,Z)$ measurement is taken, an allowance for measurement uncertainty is made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and this allowance is included in the methodology applied to the determination of the core operating limits as described in the reference in Specification 6.9.1.14.

The hot channel factors, $F_Q^M(X,Y,Z)$ and $F_{\Delta H}^M(X,Y)$, are measured periodically and compared to the nominal design values to provide a reasonable assurance that the core is operating as designed and that the limiting criteria will not be exceeded for operation within the Technical Specification limits of Sections 2.2 (Limiting Safety System Settings), 3.1.3 (Moveable Control Assemblies), 3.2.1 (AXIAL FLUX DIFFERENCE), and 3.2.4 (QUADRANT POWER TILT RATIO). An allowance is provided to account for the expected deviation between the calculation and the measurement. If the measurement is above the maximum expected value for that location, it is assumed to not be operating as designed, and a peaking margin evaluation is performed to provide a basis for decreasing the width of the AFD and f(ΔI) limits, and for reducing THERMAL POWER.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that no anomaly exists such 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 QUADRANT POWER TILT RATIO limit at which corrective action is required provides DNB and linear heat generation protection with x-y plane power tilts. The QUADRANT POWER TILT RATIO limit is reflected by a corresponding peaking augmentation factor which is included in the generation of the AFD limits.

The 2-hour time allowance for operation with the 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 $F_Q(X,Y,Z)$ is reinstated by reducing the allowable THERMAL POWER by 3 percent for each percent of tilt in excess of 1.02.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the safety analysis DNBR limit throughout each analyzed transient. R21 R130

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. R21

The flow parameters indicated in Figure 3.2-1 have been rounded down to bias the analysis in the conservative direction.

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 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 no later than the 15th of each month following the calendar month covered by the report. R64

CORE OPERATING LIMITS REPORT R146

6.9.1.14 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. $f_1(\Delta I)$ limits for Overtemperature Delta T Trip Setpoints and $f_2(\Delta I)$ limits for Overpower Delta T Trip Setpoints for Specification 2.2.1.
2. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
3. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
4. Control Bank Insertion Limits for Specification 3/4.1.3.6,
5. AXIAL FLUX DIFFERENCE Limits for Specification 3/4.2.1,
6. Heat Flux Hot Channel Factor and K(z) for Specification 3/4.2.2, and
7. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3.

6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in: R146

1. BAW-10180P-A, Rev. 1, "NEMO - NODAL EXPANSION METHOD OPTIMIZED", March 1993. (FCF Proprietary)
   (Methodology for Specification 3.1.1.3-Moderator Temperature Coefficient)
2. BAW-10169P-A, "RSG PLANT SAFETY ANALYSIS - B&W SAFETY ANALYSIS METHODOLOGY FOR RECIRCULATING STEAM GENERATOR PLANTS", October 1989. (FCF Proprietary)
   (Methodology for Specification 3.1.1.3-Moderator Temperature Coefficient)
3. BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June 1989. (FCF Proprietary)
   (Methodology for Specification 2.2.1, - Limiting Safety System Settings [$f_1(\Delta I)$, $f_2(\Delta I)$ limits], 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3/4.2.1 - Axial Flux Difference, 3/4.2.2 - Heat Flux Hot Channel Factor, 3/4.2.3 - Nuclear Enthalpy Rise Hot Channel Factor)
4. BAW-10168P-A, Rev. 2, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)
5. BAW-10168P-A, Rev 3, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (Continued)

The measurement of response time at the specified frequencies provides assurance that the protective and the engineered safety feature actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable in the updated final safety analysis report. R182

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

Action 15 of Table 3.3-1, Reactor Trip System Instrumentation, allows the breaker to be bypassed for up to 4 hours for the purpose of performing maintenance. The 4 hours is based on a Westinghouse analysis performed in WCAP-10271, Supplement 1, which determines bypass breaker availability. R46

The placing of a channel in the trip condition provides the safety function of the channel. If the channel is tripped for testing and no other condition would have indicated inoperability, the channel should not be declared inoperable. BR-10

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(X,Y,Z)$ or $F_{\Delta H}(X,Y)$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the

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MONTHLY REACTOR OPERATING REPORT

6.9.1.10 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 no later than the 15th of each month following the calendar month covered by the report. R64

CORE OPERATING LIMITS REPORT R146

6.9.1.14 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. $f_1(\Delta I)$ limits for Overtemperature Delta T Trip Setpoints and $f_2(\Delta I)$ limits for Overpower Delta T Trip Setpoints for Specification 2.2.1.
2. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
3. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
4. Control Bank Insertion Limits for Specification 3/4.1.3.6,
5. AXIAL FLUX DIFFERENCE Limits for Specification 3/4.2.1,
6. Heat Flux Hot Channel Factor and K(z) for Specification 3/4.2.2, and
7. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3.

6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in: R146

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   (Methodology for Specification 3.1.1.3-Moderator Temperature Coefficient)
2. BAW-10169P-A, "RSG PLANT SAFETY ANALYSIS - B&W SAFETY ANALYSIS METHODOLOGY FOR RECIRCULATING STEAM GENERATOR PLANTS", October 1989. (FCF Proprietary)
   (Methodology for Specification 3.1.1.3-Moderator Temperature Coefficient)
3. BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June 1989. (FCF Proprietary)
   (Methodology for Specification 2.2.1, - Limiting Safety System Settings [$f_1(\Delta I)$, $f_2(\Delta I)$ limits], 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3/4.2.1 - Axial Flux Difference, 3/4.2.2 - Heat Flux Hot Channel Factor, 3/4.2.3 - Nuclear Enthalpy Rise Hot Channel Factor)
4. BAW-10168P-A, Rev. 2, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)
5. BAW-10168P-A, Rev 3, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

CORE OPERATING LIMITS REPORT (continued)

6. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985, (W Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

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

6.9.1.14.b 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. R146

6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days after cycle start-up (Mode 2) for each reload cycle or within 30 days of issuance of any midcycle revision of the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2.1 Special reports shall be submitted within the time period specified for each report, in accordance with 10 CFR 50.4. R64

6.9.2.2 Diesel Generator Reliability Improvement Program

As a minimum the Reliability Improvement Program report for NRC audit, required by LCO 3.8.1.1, Table 4.8-1, shall include: R44

(a) a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed

(b) analysis of failures and determination of root causes of failures

(c) evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the Plant

(d) identification of all actions taken or to be taken to 1) correct the root causes of failures defined in b) above and 2) achieve a general improvement of diesel generator reliability

(e) the schedule for implementation of each action from d) above

(f) an assessment of the existing reliability of electric power to engineered-safety-feature equipment

ENCLOS'JRE 3

COMMITMENT LISTING

Commitment

TVA will revise Abnormal Operating Procedure AOP-R.05, "RCS Leak and Leak Source Identification", prior to entry into Mode 4 for Unit 1 Cycle 9 operation, to require the sampling of the Safety Injection Pump suction piping and verify that a sufficient boron concentration exists whenever leakage into the Safety Injection System from the reactor coolant system is confirmed.