INDEX

ADMINISTRATIVE CONTROLS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of the reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of reactor coolant core outlet pressure and outlet temperature has exceeded the safety limit, be in HOT STANDBY within one hour.

REACTOR CORE

2.1.2 The combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the ~~safety~~ protective limit shown in ~~Figure 2.1-2~~ the CORE OPERATING LIMITS REPORT for the various combinations of two, three and four reactor coolant pump operation.

APPLICABILITY: MODE 1.

ACTION:

Whenever the point defined by the combination of Reactor Coolant System flow, AXIAL POWER IMBALANCE and THERMAL POWER has exceeded the appropriate ~~safety~~ protective limit, be in HOT STANDBY within one hour, and comply with the requirements of Specification 6.7.2.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

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

ACTION:

MODES 1 and 2 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour.

MODES 3, 4 and 5 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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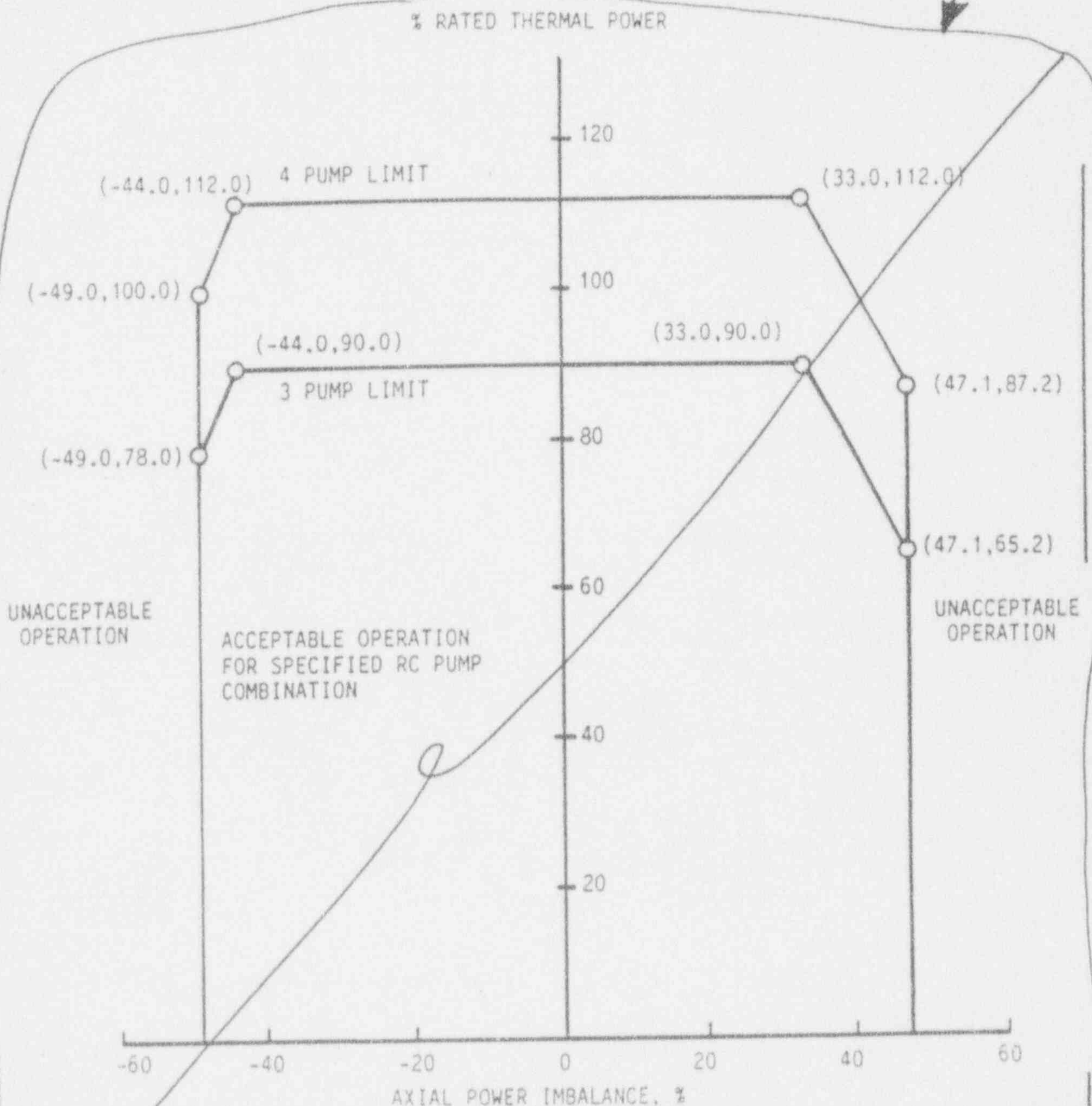
Figure 2.1-2 Reactor Core Safety Limit



Pumps Operating	Reactor Coolant Flow, gpm	Required Measured Flow to Ensure Compliance, gpm
4	380,000	389,500
3	293,860	290,957

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2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETPOINTS

2.2.1 The Reactor Protection System instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Protection System instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints

Functional unit	Trip setpoint	Allowable values
1. Manual reactor trip	Not applicable.	Not applicable.
2. High flux	≤104.94% of RATED THERMAL POWER with four pumps operating	≤104.94% of RATED THERMAL POWER with four pumps operating#
	≤80.6% of RATED THERMAL POWER with three pumps operating	≤80.6% of RATED THERMAL POWER with three pumps operating#
3. RC high temperature	≤618°F	≤618°F#
4. Flux -- Δflux/flow(1)	~~Four pump trip setpoint not to exceed the limit line of Figure 2.2-1. For three pump operation, see Figure 2.2-1~~	~~Four pump allowable values not to exceed the limit line of Figure 2.2-1#. For three pump operation, see Figure 2.2-1~~
5. RC low pressure(1)	≥1900.0 psig	≥1900.0 psig* ≥1900.0 psig**
6. RC high pressure	≤2355 psig	≤2355.0 psig* ≤2355.0 psig**
7. RC pressure-temperature(1)	≥(16.00 T_{out} °F - 7957.5) psig	≥(16.00 T_{out} °F - 7957.5) psig#
8. High flux/number of RC pumps on(1)	≤55.1% of RATED THERMAL POWER with one pump operating in each loop	≤55.1% of RATED THERMAL POWER with one pump operating in each loop#
	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop#
	≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating	≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating#
9. Containment pressure high	≤4 psig	≤4 psig#

Pump trip setpoints not to exceed the limit lines shown in the CORE OPERATING LIMITS REPORT for four and three pump operation

Pump allowable values not to exceed the limit lines shown in the CORE OPERATING LIMITS REPORT for four and three pump operation

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Table 2.2-1. (Cont'd)

(1) Trip may be manually bypassed when RCS pressure $\leq$1820 psig by actuating shutdown bypass provided that:

a. The high flux trip setpoint is $\leq$5% of RATED THERMAL POWER.

b. The shutdown bypass high pressure trip setpoint of $\leq$1820 psig is imposed.

c. The shutdown bypass is removed when RCS pressure >1820 psig.

*Allowable value for CHANNEL FUNCTIONAL TEST.

**Allowable value for CHANNEL CALIBRATION.

#Allowable value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

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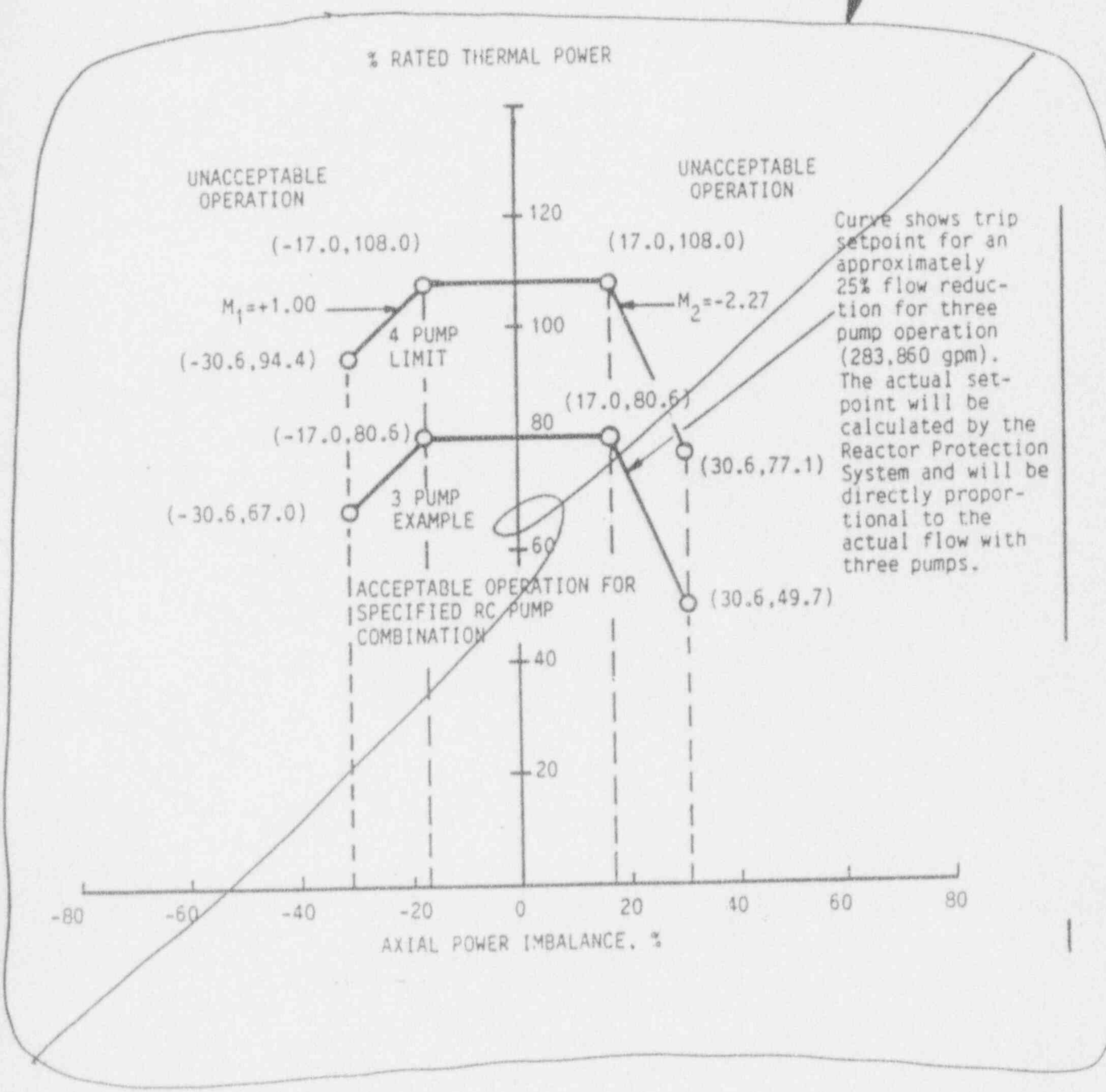
Figure 2.2-1 Trip Setpoint for Flux -- ΔFlux/Flow



2.1 SAFETY LIMITS

BASES

2.1.1 AND 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding 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 would 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 using critical heat flux (CHF) correlations. 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 B&W-2 and BWC CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The B&W-2 correlation applies to Mark-B fuel and the BWC correlation applies to all B&W fuel with zircaloy spacer grids. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30 (B&W-2) and 1.18 (BWC). The value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR equal to or greater than the correlation limit is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 380,000 GPM, which is approximately 108% of design flow rate for four operating reactor coolant pumps. (The minimum required measured flow is 389,500 GPM). This curve is based on the ~~following~~ design hot channel factors with potential fuel densification and fuel rod bowing effects~~:~~

$$\cancel{F_Q = 2.83; F^N_{\Delta H} = 1.71; F^N_Z = 1.65}$$

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod withdrawal, and form the core DNBR design basis.

The CORE OPERATING LIMITS REPORT includes curves for protective limits for AXIAL POWER IMBALANCE and for nuclear overpower based on reactor coolant system flow. A protective limit is a cycle-specific limit that ensures that a safety limit is not exceeded by requiring operation within both the cycle design (operating) limits and the Reactor Protection System setpoints. These protective limit curves reflect

SAFETY LIMITS

BASES

~~The curves of Figure 2.1-2 are based on~~ the more restrictive of two thermal limits and account for the effects of potential fuel densification and potential fuel rod bow:

design

as described in the CORE OPERATING LIMITS REPORT

1. The DNBR limit produced by a nuclear power peaking factor ~~of a~~ ~~F_Q = 2.83~~ or the combination of the radial peak, axial peak, and position of the axial peak that yields no less than the DNBR limit.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. ~~The limit is 20.5 kw/ft for all fuel in the core.~~

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for the ~~two curves of Figure 2.1-2~~ correspond to the analyzed minimum flow rates with four pumps and three pumps, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR equal to the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to the corresponding DNB correlation quality limit (+22% (B&W-2) or +26% (BWC)), whichever condition is more restrictive.

The limits for all fuel designs during the operating cycle are listed in the CORE OPERATING LIMITS REPORT.

CORE OPERATING LIMITS REPORT curves for protective limits for AXIAL POWER IMBALANCE and for nuclear overpower based on reactor coolant system flow

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SAFETY LIMITS

BASES

For the curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (B&W-2) or 1.18 (BWC) and a local quality at the point of minimum DNBR less than +22% (B&W-2) or +26% (BWC) for that particular reactor coolant pump situation. The DNBR curve for three pump operation is less restrictive than the four pump curve.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

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

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 1968 Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

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2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The reactor protection system instrumentation trip setpoints specified in Table 2.2-1 are the values at which the reactor trips are set for each parameter. The trip setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

The shutdown bypass provides for bypassing certain functions of the reactor protection system in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the shutdown bypass high pressure trip is to prevent normal operation with shutdown bypass activated. The high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The high flux trip setpoint of $\leq$5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The manual reactor trip is a redundant channel to the automatic reactor protection system instrumentation channels and provides manual reactor trip capability.

High Flux

A high flux trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches 104.94% of rated power. Due to transient overshoot, heat balance, and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

LIMITING SAFETY SYSTEM SETTINGS

BASES

RC High Temperature

The RC high temperature trip $\leq$ 618°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Flux -- ΔFlux/Flow

The power level trip setpoint produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power where protection is not provided by the high flux/number of reactor coolant pumps on trips.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Examples of typical power level and low flow rate combinations for the pump situations of Table 2.2-1 that would result in a trip are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108.0% and reactor coolant flow rate is 100% of full flow rate, or flow rate is 92.59% of full flow rate and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.68% and reactor coolant flow rate is 74.7% of full flow rate, or flow rate is 69.44% of full flow rate and power is 75%. Note that the value of 80.6% in Figure 2.2-1 was truncated from the calculated value of 80.68%.

For safety calculations the instrumentation errors for the power level were used. Full flow rate ~~in the above two examples~~ is defined as the flow calculated by the heat balance at 100% power. At the time of the calibration the RCS flow will be greater than or equal to the value in Table 3.2-2.

LIMITING SAFETY SYSTEM SETTINGS

BASES

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by a flux-to-flow ratio such that the boundaries of ~~Figure 2.2-1~~ the figure in the CORE OPERATING LIMITS REPORT are produced. 1

RC Pressure - Low, High, and Pressure Temperature

The high and low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC high pressure setpoint is reached before the high flux trip setpoint. The trip setpoint for RC high pressure, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC high pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, $\leq$ 2525 psig. The RC high pressure trip also backs up the high flux trip.

The RC low pressure, 1900.0 psig, and RC pressure-temperature ($16.00\ T_{out}$ - 7957.5) psig, trip setpoints have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

High Flux/Number of Reactor Coolant Pumps On

In conjunction with the flux - Δflux/flow trip the high flux/number of reactor coolant pumps on trip prevents the minimum core DNBR from decreasing below the minimum allowable DNB ratio by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Containment High Pressure

The Containment High Pressure Trip Setpoint $\leq$ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RC Low Pressure trip.

POWER DISTRIBUTION LIMITS

NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - F_Q

LIMITING CONDITION FOR OPERATION

within the limits specified in the CORE OPERATING LIMITS REPORT.

3.2.2 F_Q shall be ~~limited by the following relationships:~~

$$\cancel{F_Q \le \frac{2.93}{P}}$$

~~where~~ $\cancel{P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}}$ ~~and~~ $\cancel{P \le 1.0}$.

APPLICABILITY: MODE 1

ACTION:

With F_Q exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% F_Q exceeds the limit within 15 minutes and similarly reduce the high flux trip setpoint and flux-Δ flux-flow trip setpoint within 4 hours.

b. Demonstrate through incore mapping that F_Q is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that F_Q is demonstrated through incore mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.2.1 F_Q shall be determined to be within its limit by using the incore detectors to obtain a power distribution map:

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POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

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

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

c. The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured F_Q of 4.2.2.1 above, shall be increased by 1.4% to account for manufacturing tolerances and further increased by 7.5% to account for measurement uncertainty.

POWER DISTRIBUTION LIMITS

NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

within the limits specified in the CORE OPERATING LIMITS REPORT.

3.2.3 $F^N_{\Delta H}$ shall be ~~limited by the following relationship:~~

$$\cancel{F^N_{\Delta H} \le 1.71 \, [1 + 0.6(1-P)]}$$

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

$$\cancel{\text{and } P \le 1.0}$$

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% that $F^N_{\Delta H}$ exceeds the limit within 15 minutes and similarly reduce the High Flux Trip Setpoint and Flux - ΔFlux - Flow Trip Setpoint within 4 hours.

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

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POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ 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 percent of RATED THERMAL POWER after each fuel loading, and

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

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 5% for measurement uncertainty.

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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 minimum DNBR in the core $\geq$ the minimum allowable DNB ratio during normal operation and during short term transients, (b) maintaining the peak linear power density $\leq$ 18.4 kW/ft during normal operation, and (c) maintaining the peak power density less than the limits given in the bases to specification 2.1 during short term transients. In addition, the above criteria must be met in order to meet the assumptions used for the loss-of-coolant accidents.

The power imbalance envelope and the insertion limit curves defined in the CORE OPERATING LIMITS REPORT are based on LOCA analyses which have defined the maximum linear heat rate such that the maximum clad temperature will not exceed the Final Acceptance Criteria of 2200°F following a LOCA. Operation outside of the power imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The power imbalance envelope represents the boundary of operation limited by the Final Acceptance Criteria only if the control rods are at the insertion limits, as defined in the CORE OPERATING LIMITS REPORT and if the steady-state limit QUADRANT POWER TILT exists. Additional conservatism is introduced by application of:

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing tolerance factors.

e. Potential fuel rod bow effects.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensures that the original criteria are met.

The definitions of the design limit nuclear power peaking factors as used in these specifications are as follows:

F_Q Nuclear heat flux hot channel factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

POWER DISTRIBUTION LIMITS

BASES

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It has been determined by extensive analysis of possible operating power shapes that the design limits on nuclear power peaking and on minimum DNBR at full power are met ~~provided:~~ by compliance with the protective and operating limits in the CORE OPERATING LIMITS REPORT.

~~$F_Q \leq 2.93$; $F^N_{\Delta H} \leq 1.71$~~

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the AXIAL POWER IMBALANCE produced by the power peaking. It has been determined that the above hot channel factor limits will be met provided the following conditions are maintained.

1. Control rods in a single group move together with no individual rod insertion differing by more than $\pm 6.5\%$ (indicated position) from the group average height.

2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.

3. The regulating rod insertion limits of Specification 3.1.3.6 are maintained.

4. AXIAL POWER IMBALANCE limits are maintained. The AXIAL POWER IMBALANCE is a measure of the difference in power between the top and bottom halves of the core. Calculations of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with AXIAL POWER IMBALANCE. The correlation shows that the design power shape is not exceeded if the AXIAL POWER IMBALANCE is maintained between the limits specified in Specification 3.2.1.

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod insertion and are the core DNBR design basis. Therefore, for operation at a fraction of RATED THERMAL POWER, the design limits are met. When using incore detectors to make power distribution maps to determine F_Q and $F^N_{\Delta H}$:

a. The measurement of total peaking factor F_Q^{Meas}, shall be increased by 1.4 percent to account for manufacturing tolerances and further increased by 7.5 percent to account for measurement error.

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b. The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by 5 percent to account for measurement error.

For Condition II events, the core is protected from exceeding the values given in the bases to specification 2.1 locally, and from going below the minimum allowable DNB ratio by automatic protection on power, AXIAL POWER IMBALANCE pressure and temperature. Only conditions 1 through 3, above, are mandatory since the AXIAL POWER IMBALANCE is an explicit input to the reactor protection system.

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

The QUADRANT POWER TILT limit at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. In the event the tilt is not corrected, the margin for uncertainty on F_Q is reinstated by reducing the power by 2 percent for each percent of tilt in excess of the limit.

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 FSAR initial assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the minimum allowable DNB ratio throughout each analyzed transient.

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 18 month periodic measurement of the RCS total flow rate using delta P instrumentation is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 80.6% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced in accordance with Specification 2.2.1 for operation with three reactor coolant pumps operating:

   1. High Flux

   2. Flux-ΔFlux-Flow

SURVEILLANCE REQUIREMENTS

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

4.4.1.1.2 The Reactor Protection System trip setpoints for the instrumentation channels specified in the ACTION statement above shall be verified to be in accordance with Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

a. Within 4 hours after switching to a three pump combination if the switch is made while operating, or

b. Prior to reactor criticality if the switch is made while shutdown.

*See Special Test Exception 3.10.3.

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COMPOSITION

6.5.1.2 The Station Review Board (SRB) shall be composed of at least six members of the Davis-Besse onsite management organization. The members shall be as a minimum, managers or individuals reporting directly to managers from each of the following disciplines: plant operations, maintenance, planning, radiological controls, engineering, and quality assurance. The members shall meet the requirements of ANSI N18.1-1971, Sections 4.2, 4.4, or 4.6 for applicable required experience.

The SRB Chairman shall be drawn from the SRB members and designated in writing by the Plant Manager.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SRB Chairman; however, no more than two alternates shall participate as voting members in SRB activities at any one time.

MEETING FREQUENCY

6.5.1.4 The SRB shall meet at least once per calendar month and as convened by the SRB Chairman or his designee.

QUORUM

6.5.1.5 A quorum of the SRB shall consist of the Chairman or his designee and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Station Review Board shall be responsible for:

a. Review of plant administrative procedures and changes thereto.

b. Review of the safety evaluation for 1) procedures, 2) changes to procedures, equipment or systems, and 3) tests or experiments completed under the provisions of 10 CFR 50.59, to verify that such actions do not constitute an unreviewed safety question.

c. Review of proposed procedures and changes to procedures and equipment determined to involve an unreviewed safety question as defined in 10 CFR 50.59.

ADMINISTRATIVE CONTROLS

d. Review of proposed tests or experiments determined to involve an unreviewed safety question as defined in 10 CFR 50.59.

e. Review of reports of violations of codes, regulations, orders, Technical Specifications, or Operating License requirements having nuclear safety significance or reports of abnormal degradation of systems designed to contain radioactive material.

f. Review of all proposed changes to the Technical Specifications or the Operating License.

g. Deleted

h. Review of reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect plant safety.

i. Review of the Industrial Security Plan, the Security Training and Qualification Plan, and the Security Contingency Plan, and changes thereto.

j. Review of the Davis-Besse Emergency Plan and changes thereto.

k. Review of items which may constitute potential nuclear safety hazards as identified during review of facility operations.

l. Investigations or analyses of special subjects as requested by the Company Nuclear Review Board.

m. Review of all REPORTABLE EVENTS.

n. Review of all Safety Limit Violation Reports and Protective Limit Violation Reports (Section 6.7). 1

o. Review of any unplanned, accidental or uncontrolled radioactive releases, evaluation of the event, ensurance that remedial action is identified to prevent recurrence, review of a report covering the evaluation and forwarding of the report to the Plant Manager and to the CNRB.

p. Review of the changes to the OFFSITE DOSE CALCULATION MANUAL.

q. Review of the changes to the PROCESS CONTROL PROGRAM.

r. Review of the Annual Radiological Environmental Operating Report.

s. Review of the Semiannual Radioactive Effluent Release Report.

T. Review of the Fire Protection Program and changes thereto.

6.7 SAFETY LIMIT VIOLATION OR PROTECTIVE LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the NRC Operations Center by telephone as soon as possible and in all cases within one hour. In addition the Vice President, Nuclear and the CNRB shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the CNRB and the Vice President, Nuclear within 14 days of the violation.

6.7.2 INSERT attached

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Industrial Security Plan implementation.

e. Davis-Besse Emergency Plan implementation.

f. Fire Protection Program implementation.

g. The radiological environmental monitoring program.

h. The Process Control Program.

i. Offsite Dose Calculation Manual implementation.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed and approved prior to implementation as set forth in 6.5.3 above.

Insert (TS 6.7.2)

6.7.2 The following actions shall be taken in the event the Protective Limit of Specification 2.1.2 is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Protective Limit violation shall be reported to the NRC Operations Center by telephone as soon as possible and in all cases within one hour. In addition the Vice President, Nuclear and the CNRB shall be notified within 24 hours.

c. A Protective Limit Violation Report shall be prepared. The report shall be reviewed by the SRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Protective Limit Violation Report shall be submitted to the CNRB and the Vice President, Nuclear within 14 days of the violation.

microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

2.1.2 Reactor Core
2.2.1 Reactor Protection System Setpoints

6.9.1.6 Routine reports of operating statistics, shutdown experience and challenges to the Pressurizer Pilot Operated Relief Valve (PORV) and the Pressurizer Code Safety Valves shall be submitted on a monthly basis to arrive no later than the 15th of each month following the calendar month covered by the report, as follows: The signed original to the Nuclear Regulatory Commission, Document Control Desk, Washington, D. C. 20555, and one copy each to the Region III Administrator and the Davis-Besse Resident Inspector.

CORE OPERATING LIMITS REPORT

3.2.2 Nuclear Heat Flux Hot Channel Factor, F_Q
3.2.3 Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$

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

3.1.1.3c Negative Moderator Temperature Coefficient Limit
3.1.3.6 Regulating Rod Insertion Limits
3.1.3.7 Rod Program
3.1.3.8 Xenon Reactivity
3.1.3.9 Axial Power Shaping Rod Insertion Limits
3.2.1 AXIAL POWER IMBALANCE
3.2.4 QUADRANT POWER TILT

INSERT attached

~~The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be those previously reviewed and approved by the NRC, specifically:~~

~~1) BAW-10122A Rev. 1, "Normal Operating Controls," May 1984~~

~~2) BAW-10116A, "Assembly Calculations and Fitted Nuclear Data," May 1977~~

~~3) BAW-10117P-A, "Babcock & Wilcox Version of PDQ User's Manual," January 1977~~

~~4) BAW-10118A, "Core Calculational Techniques and Procedures," December 1979~~

~~5) BAW-10124A, "FLAME 3 - A Three-Dimensional Nodal Code for Calculating Core Reactivity and Power Distributions," August 1976~~

~~6) BAW-10125A, "Verification of Three-Dimensional FLAME Code," August 1976~~

~~7) BAW-10152A, "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," June 1985~~

Insert (TS 6.9.1.7)

The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be: those previously reviewed and approved by the NRC, as described in BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses", or any other new NRC-approved analytical methods used to determine core operating limits that are not yet referenced in the applicable BAW-10179P-A revision. The applicable BAW-10179P-A revision (the approved revision at the time the reload analyses are performed) shall be listed in the CORE OPERATING LIMITS REPORT. The CORE OPERATING LIMITS REPORT shall also list any new NRC-approved analytical methods used to determine core operating limits that are not yet referenced in the applicable BAW-10179P-A revision.

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

8) BAW-10119, "Power Peaking Nuclear Reliability Factors," June 1977

The methodology for Rod Program received NRC approval in the Safety Evaluation dated January 11, 1990.

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

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