LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

3.0/4.0 BASES

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: (Continued)

T'	$\leq$	588.8°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000828/psig;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

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

(1) For $q_t - q_b$ between ~~-34%~~ -25% and ~~+10.0%~~ +7.0%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds ~~-34%~~ -25%, the ΔT Trip Setpoint shall be automatically reduced by ~~2.02%~~ 2.36% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds ~~+10.0%~~ +7.0%, the ΔT Trip Setpoint shall be automatically reduced by ~~1.72%~~ 1.70% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.9% ΔT span.

Replace Existing Pages 3/4 2-1 Through 3/4 2-8
With the Attached

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

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

a. the acceptable operational space defined by Figure 3.2-1 for Relaxed Axial Offset Control (RAOC) operation, or

b. within a ± 3 percent target band about the target AFD during Base Load operation.

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

ACTION:

a. For RAOC operation with the indicated AFD outside of the Figure 3.2-1 limits, either:

   1. Restore the indicated AFD to within the Figure 3.2-1 limits within 15 minutes, or

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

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

   1. Restore the indicated AFD to within the target band limits within 15 minutes, or

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

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the Figure 3.2-1 limits.

*See Special Test Exception 3.10.2

**APL^{ND} is the minimum allowable power level for Base Load operation and will be provided in the Peaking Factor Limit Report per Specification 6.9.1.6.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the indicated AFD for each OPERABLE excore channel:

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

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

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

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

4.2.1.3 When in Base Load operation, the target AFD of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target AFD shall be updated at least once per 31 Effective Full Power Days by either determining the target AFD in conjunction with the surveillance requirements of Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

Page 3/4 2-3
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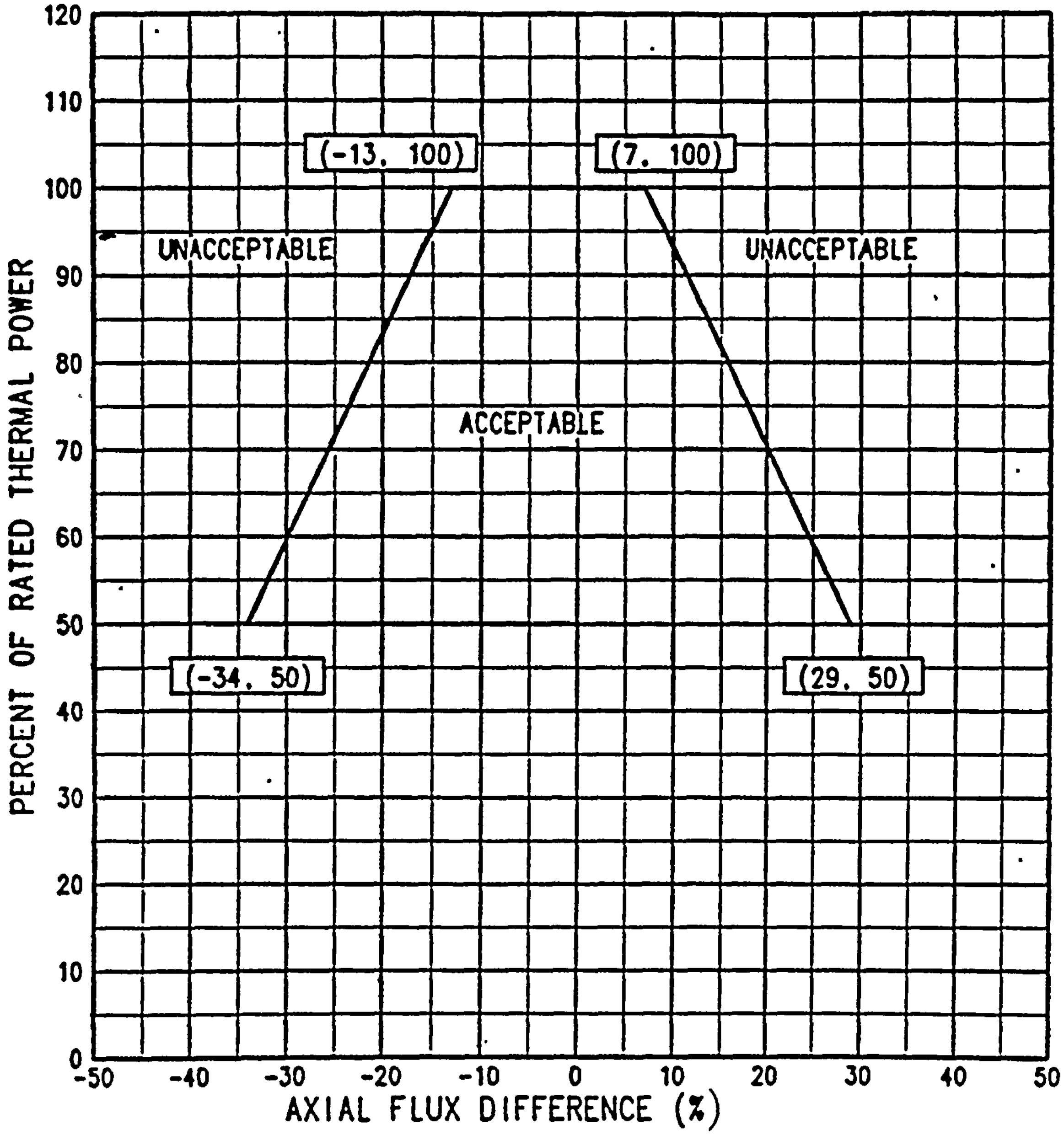

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER FOR RAOC

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z) \leq \frac{2.32}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq (4.64) [K(Z)] \text{ for } P \leq 0.5$$

Where:

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

$K(Z)$ = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

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

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{2.32}{P \times W(Z)} \times K(Z) \text{ for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{2.32}{W(Z) \times 0.5} \times K(Z) \text{ for } P \leq 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.32 is the F_Q limit, K(Z) is given in Figure 3.2-2, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6.

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

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

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

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

e. With measurements indicating

$$\text{maximum} \left(\frac{F_Q^M(Z)}{K(Z)} \right)$$

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

1) $F_Q^M(Z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c. or

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

$$\text{maximum} \left(\frac{F_Q^M(Z)}{K(Z)} \right) \text{ is not increasing.}$$

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

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

$$\left\{ \left(\text{maximum} \left[\frac{F_Q^M(Z) \times W(Z)}{\frac{2.32}{P} \times K(Z)} \right] \right) -1 \right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{ \left(\text{maximum} \left[\frac{F_Q^M(Z) \times W(Z)}{\frac{2.32}{0.5} \times K(Z)} \right] \right) -1 \right\} \times 100 \text{ for } P < 0.5$$

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits of 3.2-1 by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

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

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

SURVEILLANCE REQUIREMENTS (Continued)

g. The limits specified in Specifications 4.2.2.2c, 4.2.2.2e, and 4.2.2.2f above are not applicable in the following core plane regions:

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

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

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within $\pm$ 3% of target flux difference) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \text{minimum} \left[\frac{2.32 \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}}\right] \times 100\%$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is 2.32. K(Z) is given in Figure 3.2-2. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. The function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6.

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

4.2.2.4 During Base Load operation $F_Q(Z)$ shall be evaluated to determine if it is within its limit by:

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

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

SURVEILLANCE REQUIREMENTS (Continued)

c. Satisfying the following relationship:

$$F_Q^M(Z) \le \frac{2.32 \times K(Z)}{P \times W(Z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. The F_Q limit is 2.32.

K(Z) is given in Figure 3.2-2. P is the fraction of RATED THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6.

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

1. Prior to entering Base Load operation after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and

2. At least once per 31 effective full power days.

e. With measurements indicating

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

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

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

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

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

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

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

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

$$[(\text{max. of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{2.32}{P} \times K(Z)}]) -1] \times 100 \text{ for } P \geq APL^{ND}$$

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

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

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

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

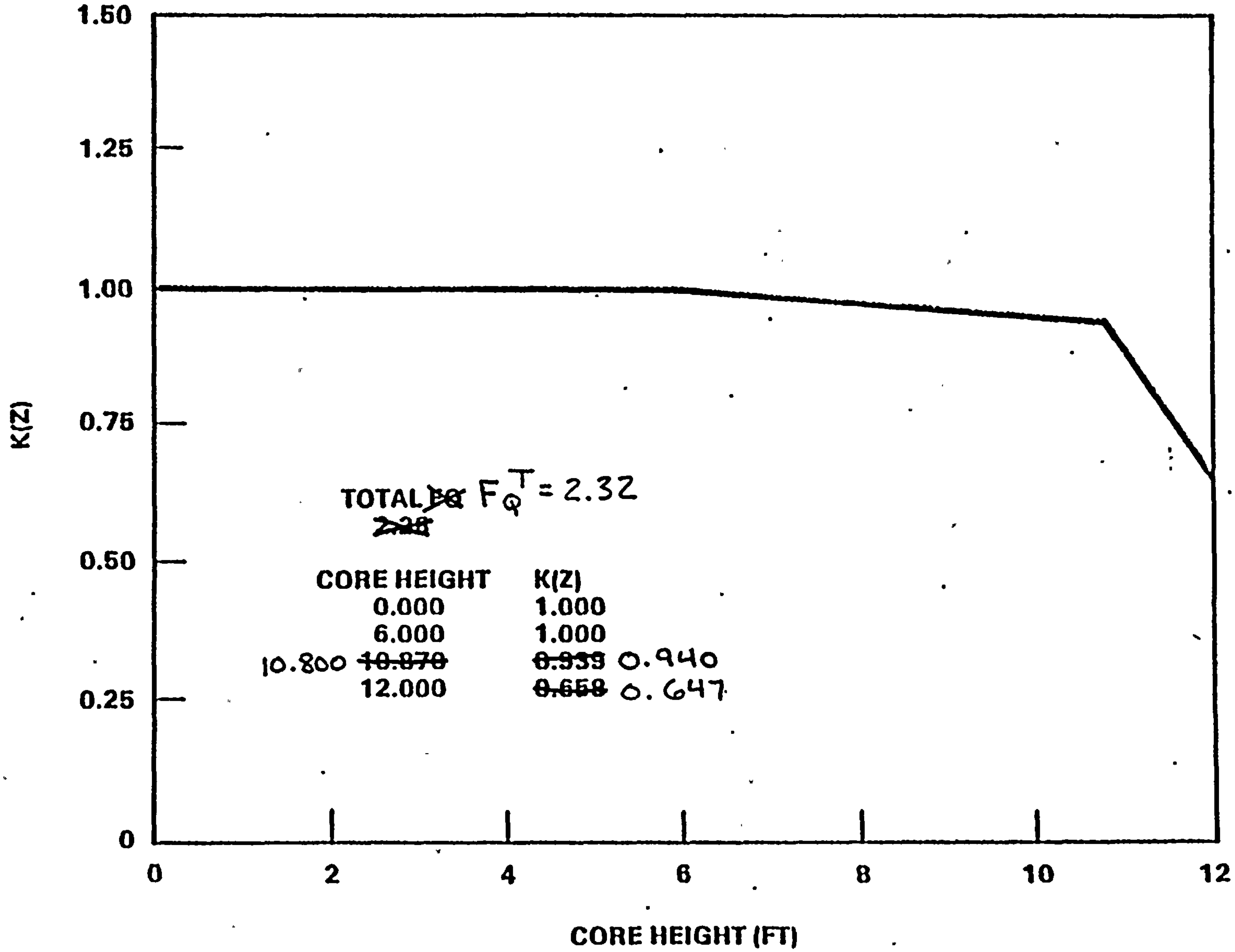

FIGURE 3.2-2

K(Z) - LOCAL AXIAL PENALTY FUNCTION FOR $F_Q(Z)$

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

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

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

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

~~$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.~~

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of ~~2.28~~ 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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

AXIAL FLUX DIFFERENCE (Continued) REPLACE WITH ATTACHED INSERT

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly t[illegible]

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

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

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

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

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

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At power levels below APL^{ND}, the limits on AFD are defined by Figure 3.2-1, i.e. that defined by the RAOC operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g. load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible; 1) RAOC, the AFD limits of which are defined by Figure 3.2-1, and 2) Base Load operation, which is defined as the maintenance of the AFD within a $\pm 3\%$ band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(Z)$ less than its limiting value. To allow operation at the maximum permissible value, the Base Load operating procedure restricts the indicated AFD to a relatively small target band and power swings (AFD target band of $\pm 3\%$, $APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24 hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are: 1) outside the allowed ΔI power operating space (for RAOC operation), or 2) outside the acceptable AFD target band (for Base Load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) APL^{ND} (for Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

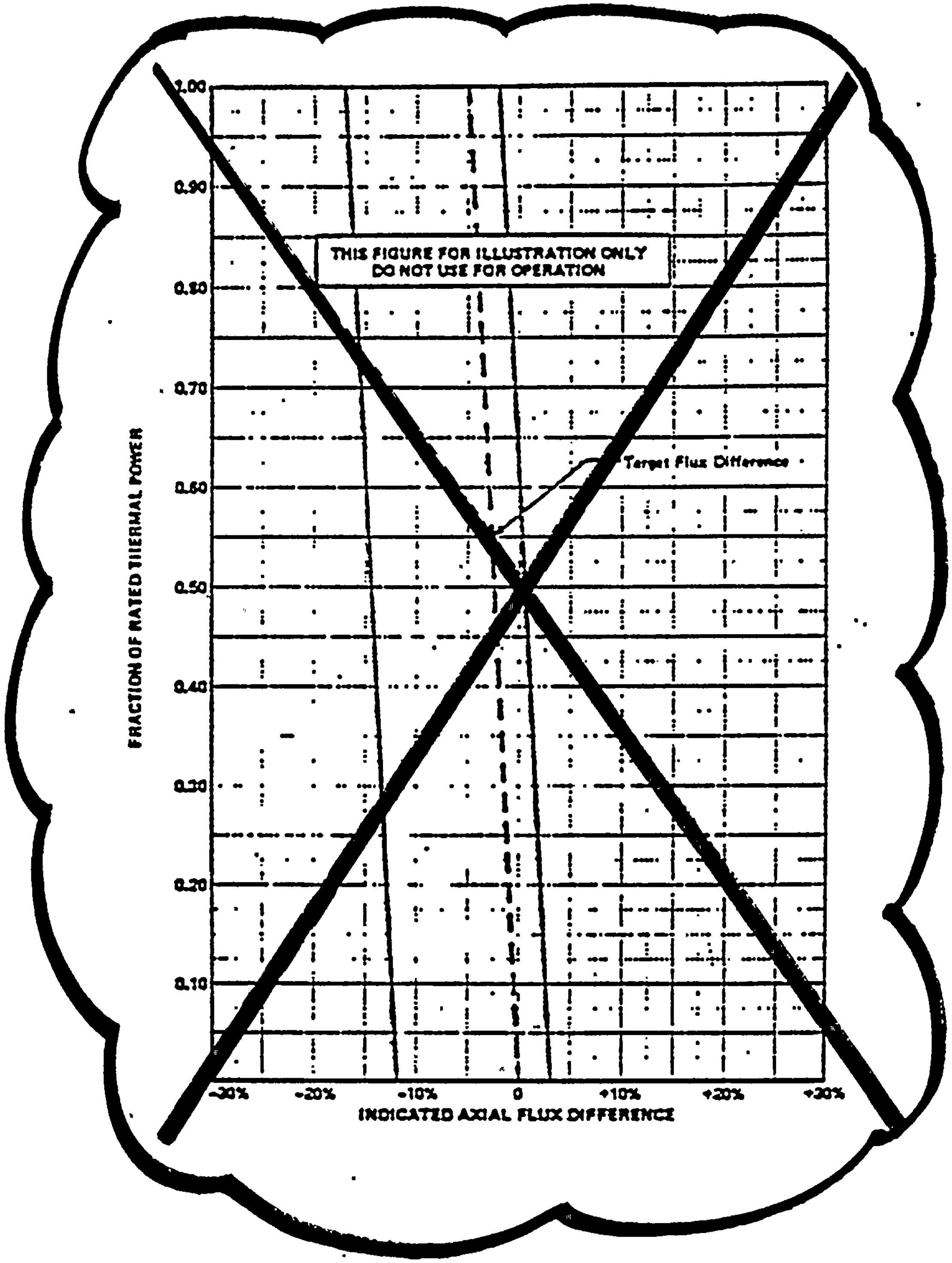

FIGURE B 3/4 2-1 (DELETED)

~~TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER FOR BURNUP GREATER THAN 6000 MWD/MTU~~

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

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

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

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The combination of the RCS flow requirement and the measurement of $F^N_{\Delta H}$ ensures that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

$F^N_{\Delta H}$ is evaluated as being less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties (less than 3% for the worst case which occurs at a burnup of 33,000 MWD/MTU). This margin includes the following:

a. Design limit DNBR of 1.30 vs 1.28,
b. Grid Spacing (K_s) of 0.046 vs 0.059,
c. Thermal Diffusion Coefficient of 0.038 vs 0.059,
d. DNBR Multiplier of 0.86 vs 0.88, and
e. Pitch reduction.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

~~The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.~~

INSERT ATTACHED

INSERT FOR PAGE B 3/4 2-4

The hot channel factor $F_Q^M(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, W(Z) or $W(Z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(Z)$, is met. W(Z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(Z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. The W(Z) function for normal operation is provided in the Peaking Factor Limit Report per Specification 6.9.1.6.

(008CRS)

~~RADIAL~~ PEAKING FACTOR LIMIT REPORT   REPLACE WITH ATTACHED INSERT

~~6.9.1.6 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the NRC Regional Administrator with a copy to the Director of Nuclear Reactor Regulation, Attention: Chief, Reactor Systems Branch, Division of PWR Licensing-A, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, for all core planes containing Bank "D" control rods and all unrodded core planes and the plot of predicted ($F_q^T \cdot P_{Rel}$) vs. Axial Core Height with the limit envelope at least 60 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter. In addition, in the event that the limit should change requiring a new substantial or an amended submittal to the Radial Peaking Factor Limit Report, it will be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.~~

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

Insert For Page 6-24

The W(Z) Functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be established for each reload core and implemented prior to use.

The methodology used to generate the W(Z) functions for RAOC and Base Load operation and the value for APL^{ND} shall be those previously reviewed and approved by the NRC.* If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the W(Z) functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be provided to the NRC in accordance with 10 CFR 50.4 within 30 days after each cycle initial criticality.

Any information needed to support W(Z), $W(Z)_{BL}$, and APL^{ND} will be by request from the NRC and need not be included in this report.

* WCAP-10216 "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification".

ENCLOSURE 3
SECTION 1

SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400
OPERATING LICENSE NPF-63
REQUEST FOR LICENSE AMENDMENT

BASIS FOR CHANGE REQUEST
INCREASED F-DELTA-H MULTIPLIER

Proposed Change

The proposed amendment revises the equation used to determine F-Delta-H, the Nuclear Enthalpy Rise Hot Channel Factor, presented in Technical Specification 3.2.3.b. The existing 0.2 multiplier has been increased to 0.3. This multiplier acts to increase the allowable F-Delta-H at reduced power levels. In addition, the core limit curves presented in Technical Specification Figure 2.1-1 have been revised. These curves, which are based on F-Delta-H, show the loci of points of thermal power, Reactor Coolant System pressure and average temperature for which the minimum Departure from Nucleate Boiling Ratio (DNBR) is no less than 1.30. Similar changes to the F-Delta-H multiplier have been granted for McGuire Units 1 and 2 and Catawba Units 1 and 2.

Basis

F-Delta-H, the 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. It represents the allowable extent of radial power peaking. Technical Specification 3.2.3.b limits F-Delta-H by the following function:

$$\text{F-Delta-H} \leq 1.49 \times [1 + 0.2(1 - P)]$$

where P is the fraction of rated thermal power. The proposed amendment replaces the existing 0.2 multiplier with 0.3. Increasing the multiplier from 0.2 to 0.3 increases loading pattern flexibility and allows deeper rod insertion at low power levels. As a result of the increased F-Delta-H multiplier the core limit curves presented in Technical Specification Figure 2.1-1 have been revised. These curves, which are based on F-Delta-H, show the loci of points of thermal power, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30.

The effect of changes in the F-Delta-H multiplier and the associated core limit curves on the safety analyses presented in the SHNPP FSAR has been evaluated. The full power value for F-Delta-H remains the same; however, transients initiated from reduced power which directly model F-Delta-H will be affected by the changes. Indirectly, any events which

rely on the OTΔT/OPΔT setpoints for protection may be impacted by the increase in the F-Delta-H multiplier. Since a power dependent value of F-Delta-H is assumed in the generation of the core limits, the increase in F-Delta-H at reduced power will result in a change to the core limits below 100% power. The existing OTΔT/OPΔT setpoints have been compared to the revised core limits which include the F-Delta-H multiplier increase. It was determined that the existing setpoints continue to protect the core limits, and thus events which rely on the OTΔT/OPΔT setpoints for protection are not impacted by this change. Therefore, the conclusions of the non-LOCA analyses presented in the SHNPP FSAR remain valid.

The worst case LOCA analyses presented in the SHNPP FSAR assume operation at 100% power. Any LOCA event initiated at lower reactor power is less limiting. With the reduced power required to cause a change in F-Delta-H, the linear heat rate at locations on the hot rod in the hot assembly affected by F-Delta-H would remain lower than the linear heat rate assumed in the FSAR analyses. Therefore the LOCA analyses presented in the SHNPP FSAR continue to bound the F-Delta-H multiplier increase and remain valid.

ENCLOSURE 3
SECTION 2

SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400
OPERATING LICENSE NPF-63
REQUEST FOR LICENSE AMENDMENT

10CFR50.92 EVALUATION
INCREASED F-DELTA-H MULTIPLIER

The Commission has provided standards in 10CFR50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed this proposed license amendment request and determined that its adoption would not involve a significant hazards consideration. The bases for this determination are as follows:

Proposed Change

The proposed amendment revises the equation used to determine F-Delta-H, the Nuclear Enthalpy Rise Hot Channel Factor, presented in Technical Specification 3.2.3.b. The existing 0.2 multiplier has been increased to 0.3. This multiplier acts to increase the allowable F-Delta-H at reduced power levels. In addition, the core limit curves presented in Technical Specification Figure 2.1-1 have been revised. These curves, which are based on F-Delta-H, show the loci of points of thermal power, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30.

Basis

The change does not involve a significant hazards consideration for the following reasons:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed amendment does not alter the method in which any safety related system performs its intended function. As such, this change does not increase the probability of any previously evaluated accidents. The FSAR accident analyses were reviewed and it was determined that the conclusions in the FSAR

remain valid for the proposed increase in the F-Delta-H multiplier. Non-LOCA transients initiated from reduced power which directly model F-Delta-H are affected by the changes. Indirectly, any events which rely on the OTΔT/OPΔT setpoints for protection may be impacted by the increase in the F-Delta-H multiplier. Since a power dependent value of F-Delta-H is assumed in the generation of the core limits, the increase in F-Delta-H at reduced power will result in a change to the core limits below 100% power. The existing OTΔT/OPΔT setpoints were compared to the revised core limits which include the F-Delta-H multiplier increase. It was determined that the existing setpoints continue to protect the core limits, and thus events which rely on the OTΔT/OPΔT setpoints for protection remain within their acceptance criteria.

The worst case LOCA analyses presented in the SHNPP FSAR assume operation at 100% power. Any LOCA event initiated at lower reactor power is less limiting. With the reduced power required to cause a change in F-Delta-H, the linear heat rate at locations on the hot rod in the hot assembly affected by F-Delta-H would remain lower than the linear heat rate assumed in the FSAR analyses. Therefore, the LOCA analyses presented in the SHNPP FSAR continue to bound the F-Delta-H multiplier increase and remain valid.

Based on the above, the consequences of previously evaluated accidents are not significantly increased.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. As stated in Item 1, the proposed amendment does not introduce any new equipment or require any existing equipment or systems to perform a different type of function than they are currently designed to perform. Therefore, the proposed change can not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety. The existing SHNPP FSAR LOCA and non-LOCA analyses have been evaluated and the conclusions remain valid with the increased F-Delta-H multiplier. As such, the change does not involve a significant reduction in the margin of safety.

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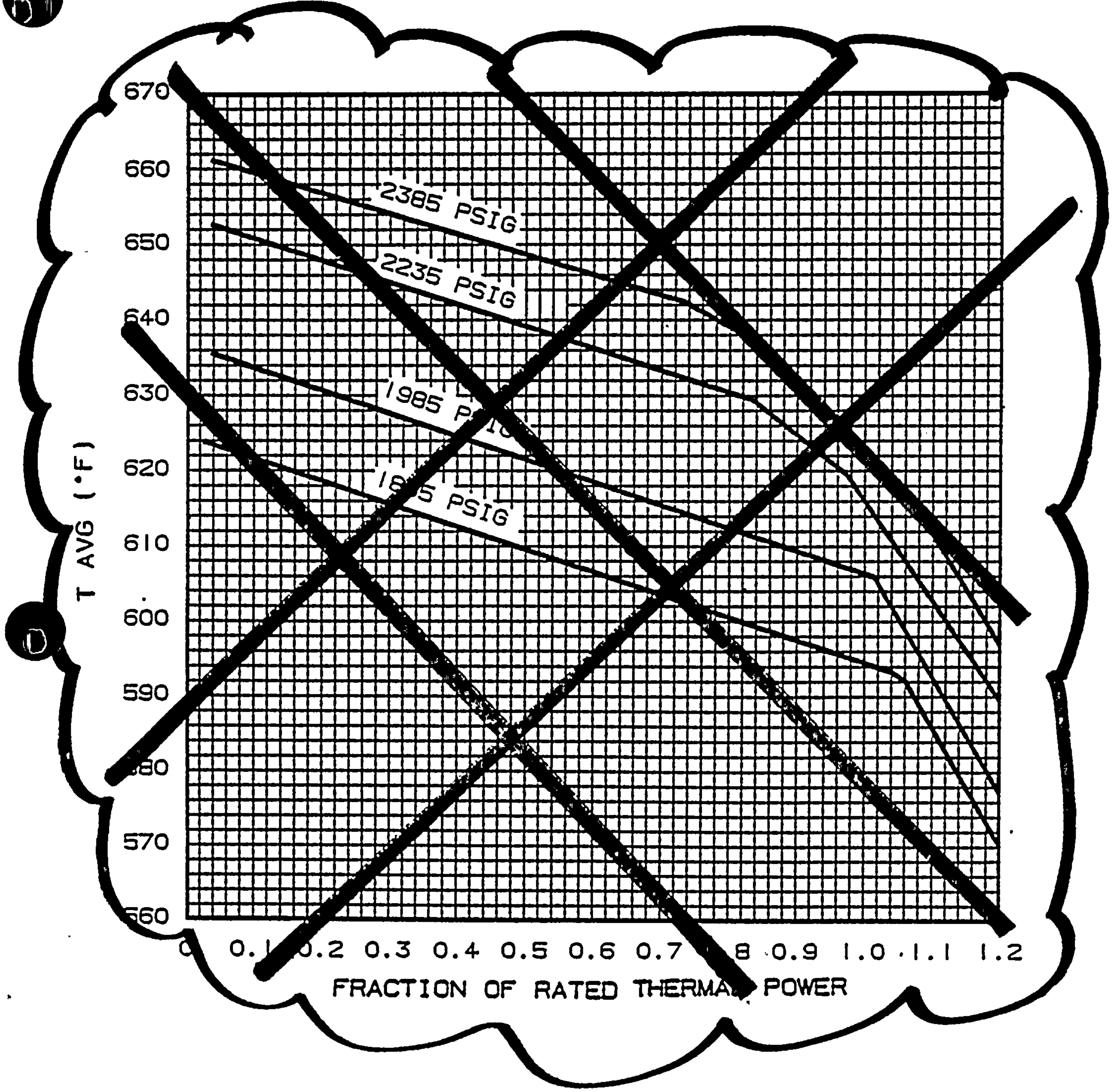

FIGURE 2.1-1

REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION

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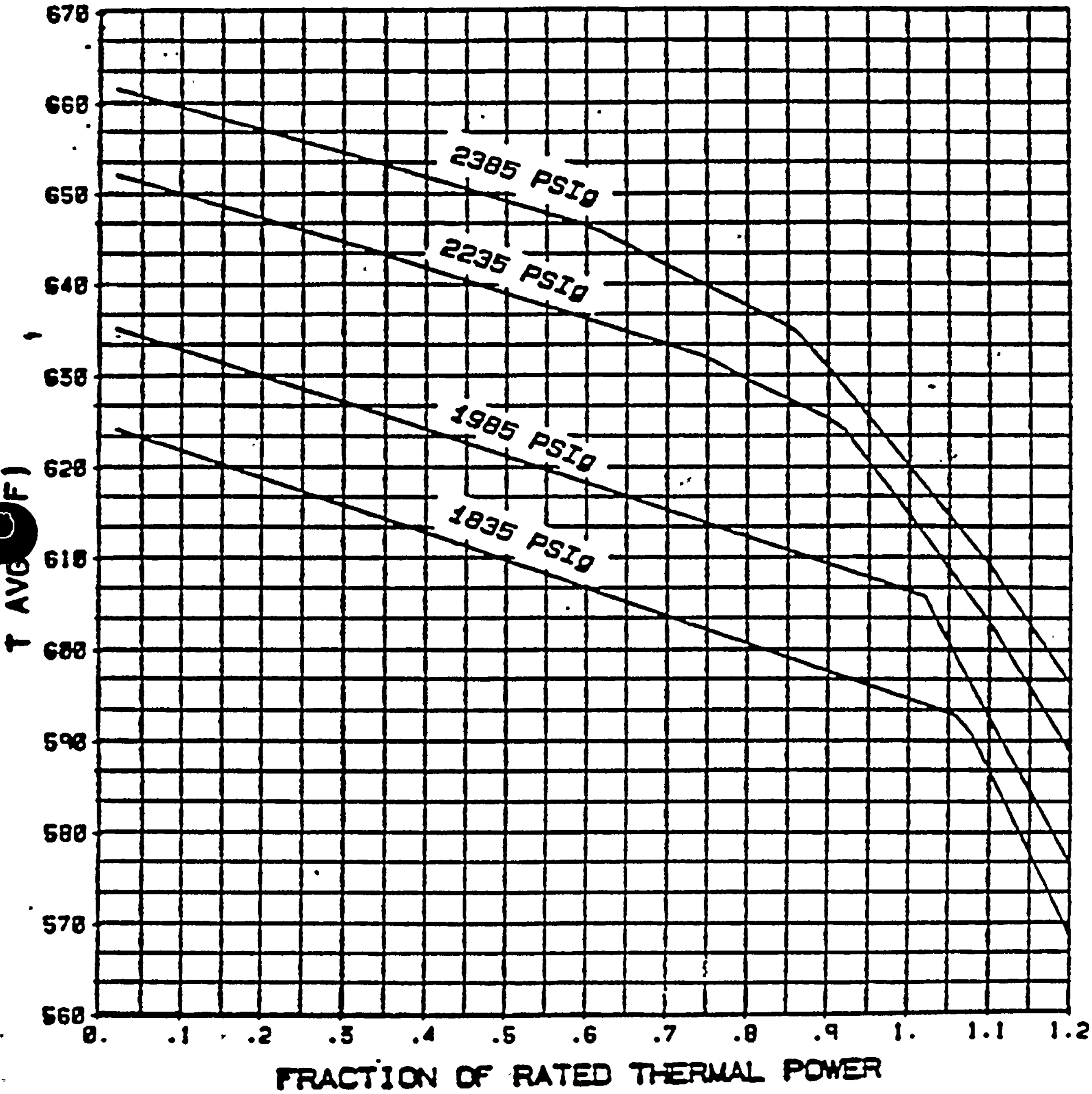

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the calculated heat flux that would cause DNB at a particular core location to the actual local heat flux and is indicative of the margin to DNB.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

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 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

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

$$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

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

LIMITING CONDITION FOR OPERATION

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

a. Measured RCS flow rate $\geq$ 292,800 gpm x $(1.0 + C_1)$, and

b. Measured $F^N_{\Delta H} \leq 1.49\,[1.0 + \cancel{0.2}(1.0-P)]$
0.3

Where:

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

$F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map, and the measured values of $F^N_{\Delta H}$ shall be used for comparison above since the 1.49 value above accounts for a 4% allowance on incore measurement of $F^N_{\Delta H}$.

C_1 = Measurement uncertainty for core flow as described in the Bases.

APPLICABILITY: MODE 1.

ACTION:

With RCS total flow rate or $F^N_{\Delta H}$ outside the above limits:

a. Within 2 hours either:

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

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

ENCLOSURE 4
SECTION 1

SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400
OPERATING LICENSE NPF-63
REQUEST FOR LICENSE AMENDMENT

BASIS FOR CHANGE REQUEST
MISCELLANEOUS TECHNICAL SPECIFICATION CHANGES

Proposed Change

The proposed amendment revises: (1) the Technical Specification Table 4.3-1 Surveillance Requirements for the Excore Power Range Monitors; and (2) the description of Fuel Assemblies located in Technical Specification 5.3.1. Currently, Technical Specification Table 4.3-1 requires that a single-point comparison of INCORE/EXCORE Axial Flux Difference (AFD) be performed monthly and an INCORE/EXCORE calibration be performed quarterly. The proposed change to Table 4.3-1 revises these surveillances to once per 31 Effective Full Power Days (EFPD) and once per 92 EFPD respectively.

The proposed change to the fuel assembly description of Technical Specification 5.3.1 allows for repair of fuel assemblies by substitution of filler rods or vacancies for damaged fuel rods.

Basis

The proposed change to Technical Specification Table 4.3-1 revises the calibration surveillance frequencies for the Excore detectors from a calendar day basis to an EFPD basis. One of the functions of the excore power range monitors is to provide indication of the core AFD. Recalibration of these channels is periodically required to assure that the AFD determined by the channels matches the incore AFD. The need for recalibration arises due to changes in core power distribution caused by fuel and burnable poison depletion. This depletion is a function of core burnup, not merely time. As such, a surveillance interval based on EFPD is more appropriate. In addition, the single-point calibration check described in Note 3 to Technical Specification Table 4.3-1 is typically performed by comparing the AFD predicted by a full core flux map to that indicated by the power range channels. Since Technical Specification 3/4.2.3 requires a full core flux map every 31 EFPD, the proposed surveillance interval revision provides consistency and minimizes the duty on the incore detector system.

The proposed change to the fuel assembly description in Technical Specification 5.3.1 would allow for replacement of damaged fuel rods with filler rods or vacancies in a fuel assembly provided the replacement is justified by a cycle specific evaluation. Identification and replacement of damaged fuel rods could extend the affected

assemblies life. A cycle specific evaluation with respect to the affect on safety will be performed prior to the use of the repaired assembly in the core. This evaluation would address such things as the affect on core thermal limits and overall core reactivity as well as any mechanical considerations.

ENCLOSURE 4
SECTION 2

SHEARON HARRIS NUCLEAR POWER PLANT-
NRC DOCKET NO. 50-400
OPERATING LICENSE NPF-63
REQUEST FOR LICENSE AMENDMENT

10CFR50.92 EVALUATION
MISCELLANEOUS TECHNICAL SPECIFICATION CHANGES

The Commission has provided standards in 10CFR50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed this proposed license amendment request and determined that its adoption would not involve a significant hazards consideration. The bases for this determination are as follows:

Proposed Change

The proposed amendment revises: (1) the Technical Specification Table 4.3-1 Surveillance Requirements for the Excore Power Range Monitors; and (2) the description of Fuel Assemblies located in Technical Specification 5.3.1. Currently, Technical Specification Table 4.3-1 requires that a single-point comparison of INCORE/EXCORE Axial Flux Difference (AFD) be performed monthly and an INCORE/EXCORE calibration be performed quarterly. The proposed change to Table 4.3-1 revises these surveillances to once per 31 Effective Full Power Days (EFPD) and once per 92 EFPD respectively.

The proposed change to the fuel assembly description of Technical Specification 5.3.1 allows for repair of fuel assemblies by substitution of filler rods or vacancies for damaged fuel rods.

Basis

The change does not involve a significant hazards consideration for the following reasons:

1. Revising the surveillance frequencies associated with calibration of the excore detectors from a strictly calendar basis to an Effective Full Power Days basis does not affect any system or component involved in the mitigation of an event. The change to the surveillance interval does not alter the channels ability to

perform their necessary functions. The instruments response changes as a function of core exposure and is not dependent on the number of calendar days between surveillance. Therefore, the proposed revision to the Technical Specification Table 4.3-1 surveillance frequencies does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the fuel assembly description in Technical Specification 5.3.1 does not affect any system or component involved in the mitigation of an event. The change merely allows replacement of damaged fuel rods with filler rods or vacancies provided a cycle specific evaluation is performed to justify the modification. Therefore, the proposed revision to Technical Specification 5.3.1 does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes to Technical Specification Table 4.3-1 and Technical Specification 5.3.1 do not require the use of a new or different system than currently exists, nor do they require existing systems to perform functions for which they were not intended. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety. The change to the surveillance intervals associated with the calibration of the excore power range channels specified in Technical Specification Table 4.3-1 does not adversely affect their operability. The instruments response changes as a function of core exposure and is not dependent on the number of calendar days between surveillance. Therefore, this change does not result in a significant reduction in the margin of safety.

   The proposed change to Technical Specification 5.3.1 allowing substitution of filler rods or vacancies for damaged fuel rods requires that the modification be analyzed prior to implementation. The evaluation will take into account the actual number and location of the filler rods. The configuration used will be such that acceptance criterion presented in the SHNPP FSAR will continue to be met, thereby assuring that there will be no reduction in the margin of safety.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(12)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), # ~~M~~(3, 4), ## ~~Q~~(4, 6), R(4, 5)	Q(15)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q(15)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q(15)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1), Q(8, 15)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	S	R(11)	Q(15)	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q(15)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(15)	N.A.	N.A.	1 (16)
10. Pressurizer Pressure--High	S	R	Q(15)	N.A.	N.A.	1, 2

[illegible]

Each 31 Effective Full Power Days
Each 92 Effective Full Power Days

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

***Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(9) Setpoint verification is not applicable.

(10) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the reactor trip breakers.

(11) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

(12) Verify that appropriate signals reach the undervoltage and shunt trip relays, for both the main and bypass breakers, from the manual reactor trip switch.

DESIGN FEATURES

except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, stainless steel, or by vacancies may be made if justified by a cycle specific evaluation.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly Normally containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.9 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 9410 ± 100 cubic feet at a nominal T_{avg} of 588.8°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological station shall be located as shown on Figure 5.1-1.

ENCLOSURE 5

SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400
OPERATING LICENSE NPF-63
BASES CHANGE

BASIS CHANGE - ROD BOW PENALTY

Description of Change

This change revises Bases Section 3/4.2.2 and 3/4.2.3, "Heat Flux Hot Channel Factor, RCS Flow Rate, and Nuclear Enthalpy Rise Hot Channel Factor". The parenthetical note on page B 3/4 2-4 containing the value (and associated burnup level) of DNBR penalty due to rod bow has been replaced with a reference to Section 4.4.2.2.4.2 of the FSAR, which contains this and other DNBR penalties. This change to the bases is similar to that for other operating units such as V. C. Summer and Wolf Creek.

Since the change affects only the Bases of the SHNPP Technical Specifications, which are not license requirements, a Significant Hazards Analysis, notice for public comment in the Federal Register, and prior NRC approval are not required. CP&L has evaluated this change in accordance with 10 CFR 50.59 and determined that an unreviewed safety question does not exist. The Bases change will become effective at the time of initial criticality for SHNPP Cycle 2.

Basis

The change deletes the discussion of allocated generic DNBR margin to offset the DNBR penalty due to rod bow in Bases Section 3/4.2.2 and 3/4.2.3 and in its place references the SHNPP FSAR section containing rod bow and other DNBR penalties. This is consistent with the SER which was issued with the SHNPP Operating License on January 12, 1987. Page 14 of the SER states in part: "Documentation in the FSAR of the use of the generic DNBR margin is also acceptable." Removing the rod bow DNBR penalty from the Bases will eliminate the need for future Bases changes and avoid possible operator confusion caused by having an outdated value in the Technical Specification Bases.

The phenomenon of fuel rod bowing must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. For SHNPP Cycle 2 operation, the maximum value for rod bow penalty will be changed from < 3% at a burnup of 33,000 MWD/MTU to < 1.5% at 24,000 MWD/MTU. This lower maximum penalty represents an analysis refinement by Westinghouse that slightly reduces the calculated consequences of malfunctions or accidents. This decrease has been generically approved and accepted by the NRC in Reference 1.

References

1. Letter from C. Berlinger (NRC) to E. P. Rahe, Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 1986.

The applicable value of rod bow and any other penalties is presented in FSAR Section 4.4.2.2.4.2.

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

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

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

$F_{\Delta H}^{N}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The combination of the RCS flow requirement and the measurement of $F_{\Delta H}^{N}$ ensures that the calculated DNBR will not be below the design DNBR value. The relaxation of $F_{\Delta H}^{N}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

$F_{\Delta H}^{N}$ is evaluated as being less than or equal to 1.49. This value is used in the various accident analyses where $F_{\Delta H}^{N}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. ~~(less than 3% for the worst case which occurs at a burnup of 33,000 MWD/MTU)~~. This margin includes the following:

a. Design limit DNBR of 1.30 vs 1.28,
b. Grid Spacing (K_s) of 0.046 vs 0.059,
c. Thermal Diffusion Coefficient of 0.038 vs 0.059,
d. DNBR Multiplier of 0.86 vs 0.88, and
e. Pitch reduction.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

ENCLOSURE 6

SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400
OPERATING LICENSE NPF-63
REQUEST FOR LICENSE AMENDMENT

<u>COMPREHENSIVE PACKAGE OF
TECHNICAL SPECIFICATION PAGES</u>

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

3.0/4.0 BASES

SECTION | PAGE

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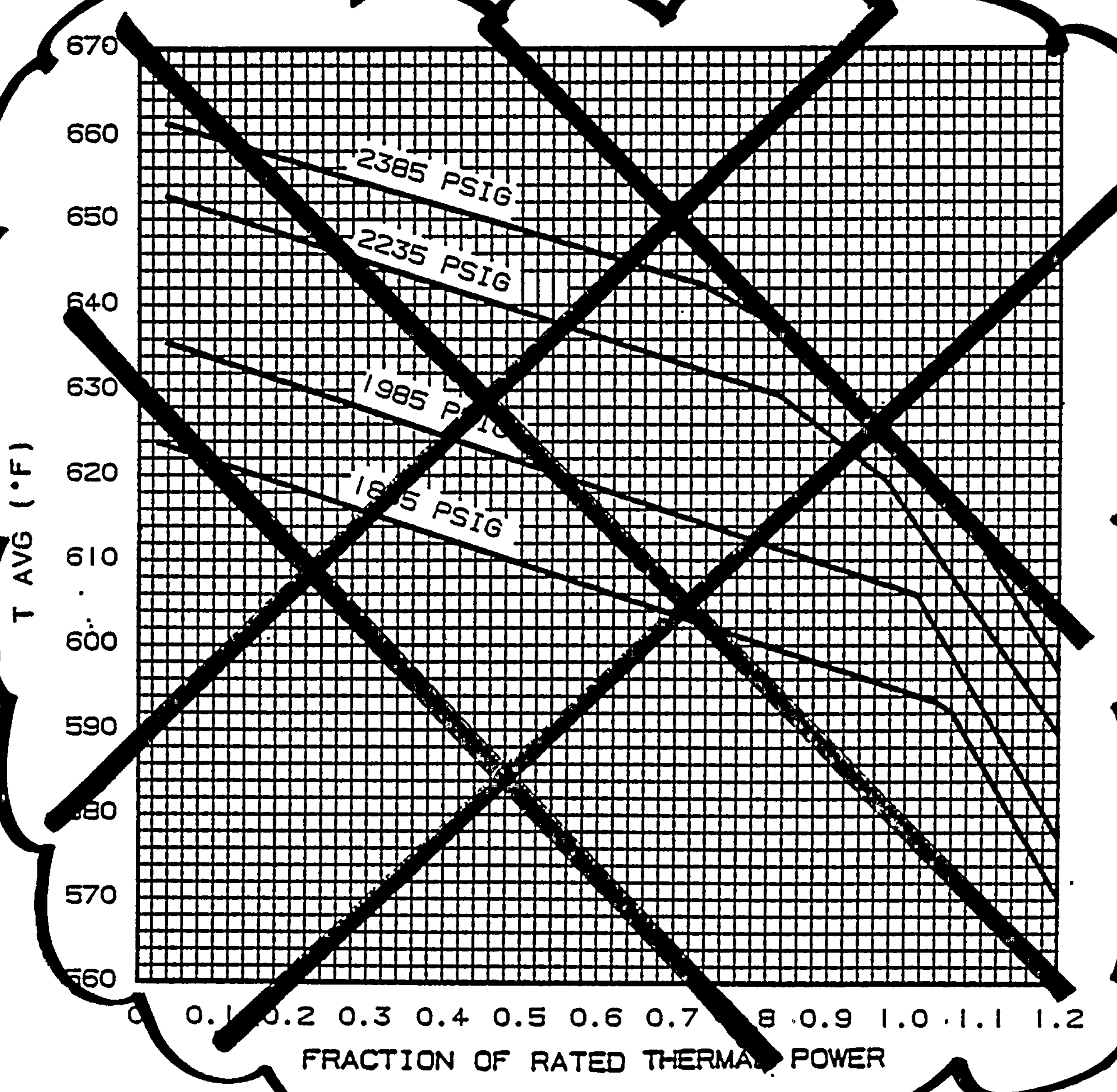

FIGURE 2.1-1

REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION

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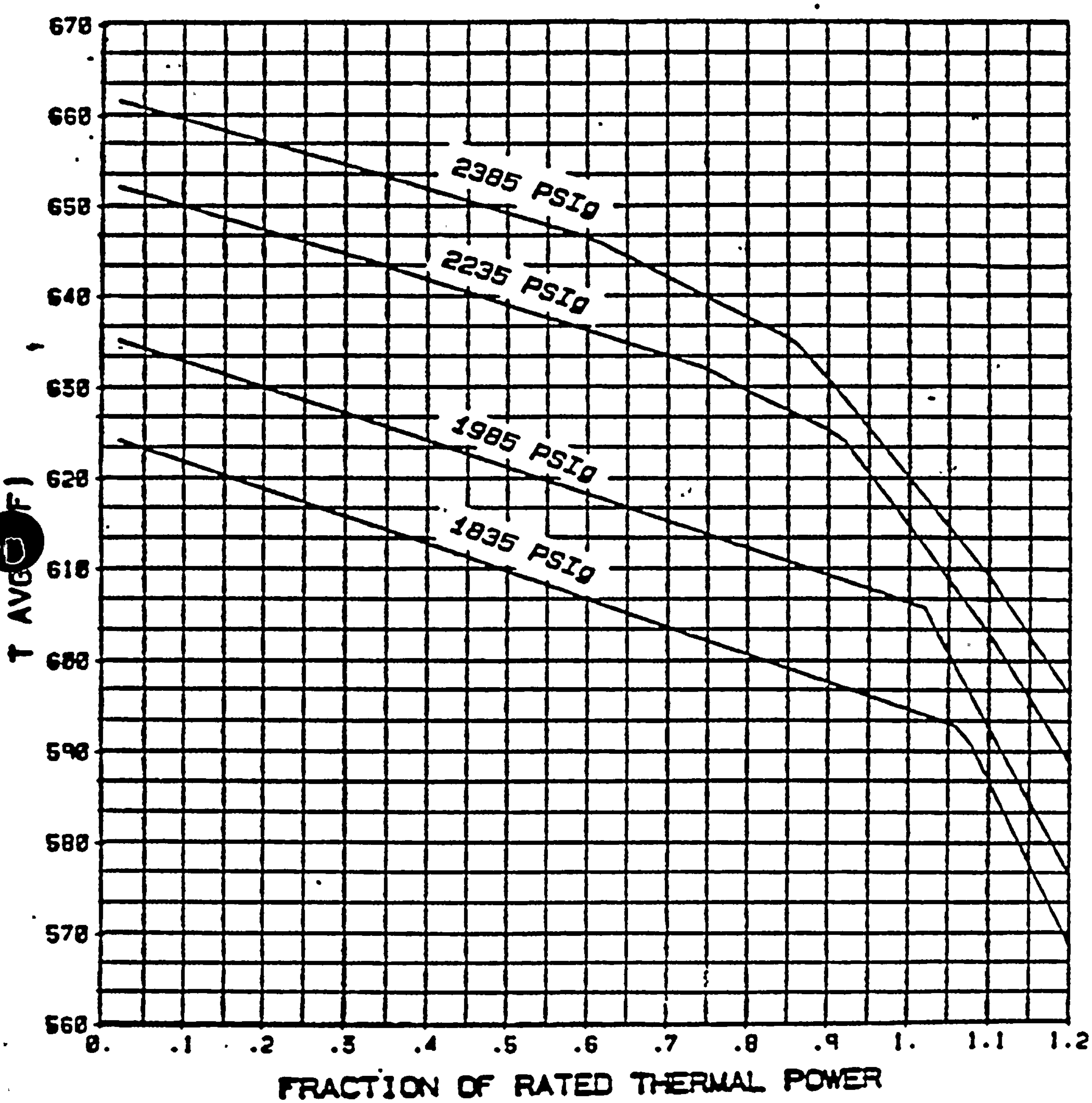

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: (Continued)

T'	$\leq$	588.8°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000828/psig;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

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

(1) For $q_t - q_b$ between ~~-34%~~ -25% and ~~+10.0%~~ +7.0%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds ~~-34%~~ -25%, the ΔT Trip Setpoint shall be automatically reduced by ~~2.02%~~ 2.36% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds ~~+10.0%~~ +7.0%, the ΔT Trip Setpoint shall be automatically reduced by ~~1.72%~~ 1.70% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.9% ΔT span.

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 through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the calculated heat flux that would cause DNB at a particular core location to the actual local heat flux and is indicative of the margin to DNB.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

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 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

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

$$F^N_{\Delta H} = 1.55\ [1+ \cancel{0.2}\ 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

REPLACE WITH ATTACHED

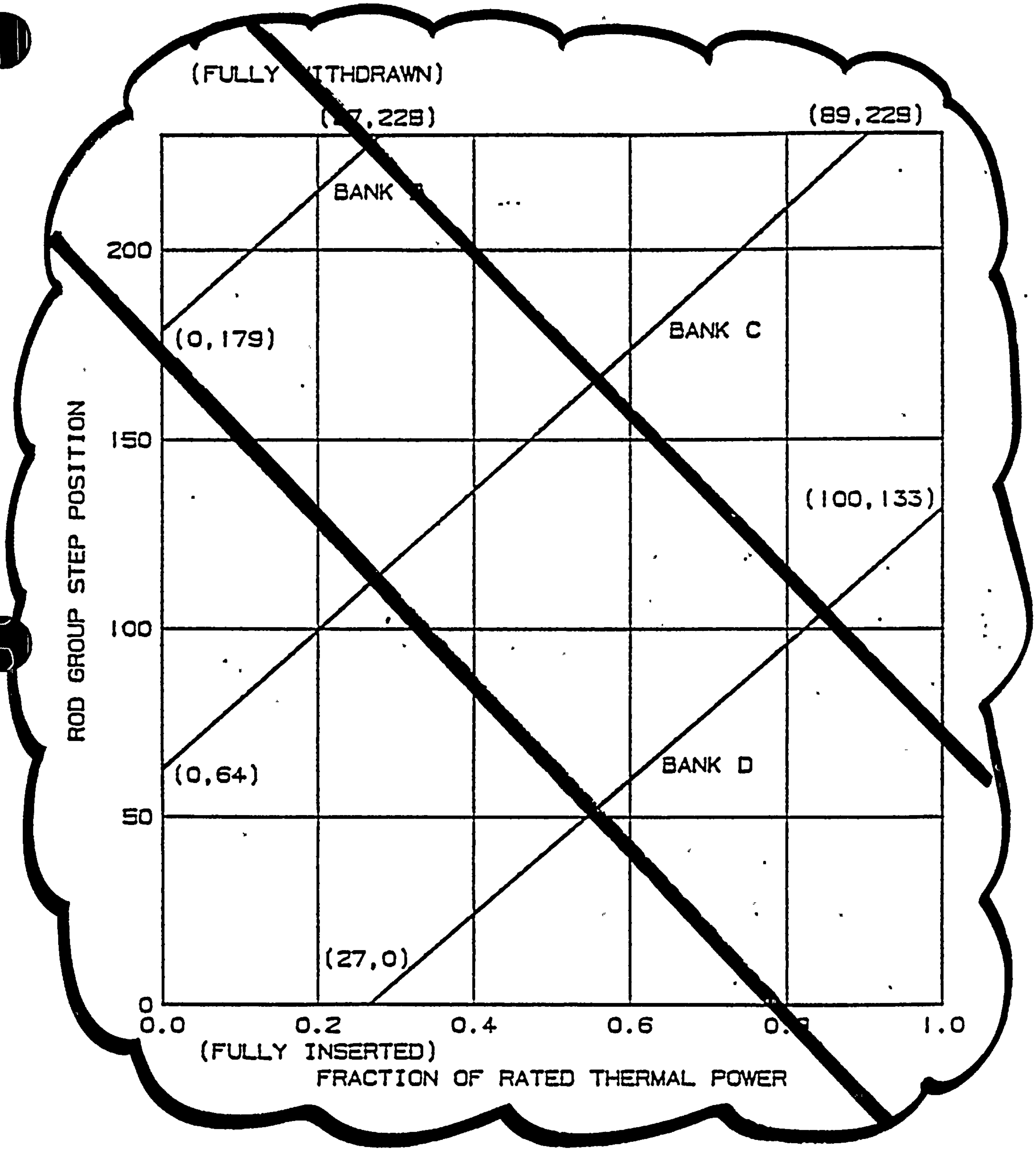

FIGURE 3.1-1

ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER THREE-LOOP OPERATION

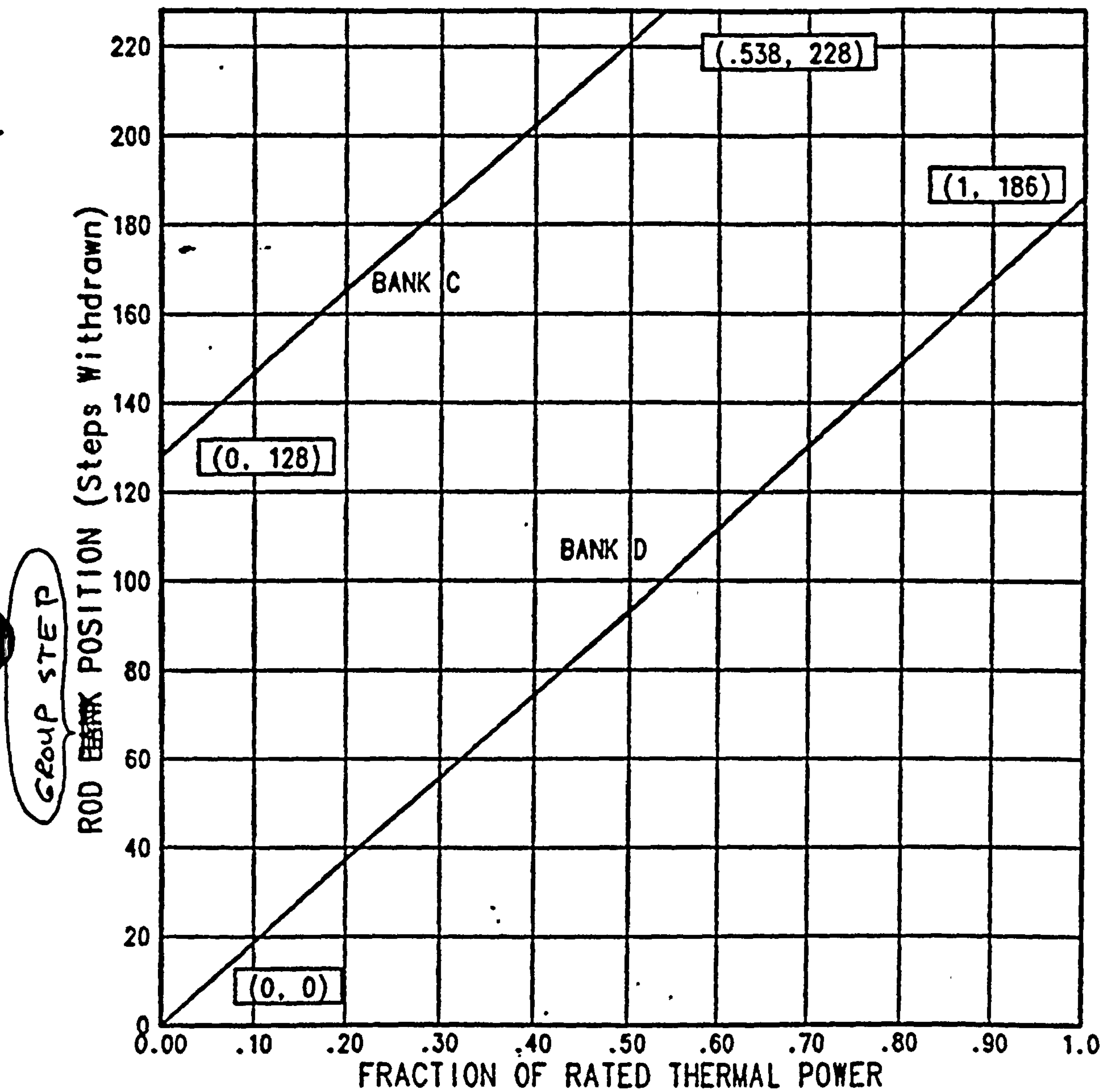

FIGURE 3.1-1

ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER

THREE-LOOP OPERATION

Replace Existing Pages 3/4 2-1 Through 3/4 2-8
With the Attached

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

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

a. the acceptable operational space defined by Figure 3.2-1 for Relaxed Axial Offset Control (RAOC) operation, or

b. within a ± 3 percent target band about the target AFD during Base Load operation.

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

ACTION:

a. For RAOC operation with the indicated AFD outside of the Figure 3.2-1 limits, either:

   1. Restore the indicated AFD to within the Figure 3.2-1 limits within 15 minutes, or

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

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

   1. Restore the indicated AFD to within the target band limits within 15 minutes, or

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

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the Figure 3.2-1 limits.

*See Special Test Exception 3.10.2

**APL^{ND} is the minimum allowable power level for Base Load operation and will be provided in the Peaking Factor Limit Report per Specification 6.9.1.6.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the indicated AFD for each OPERABLE excore channel:

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

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

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

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

4.2.1.3 When in Base Load operation, the target AFD of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target AFD shall be updated at least once per 31 Effective Full Power Days by either determining the target AFD in conjunction with the surveillance requirements of Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

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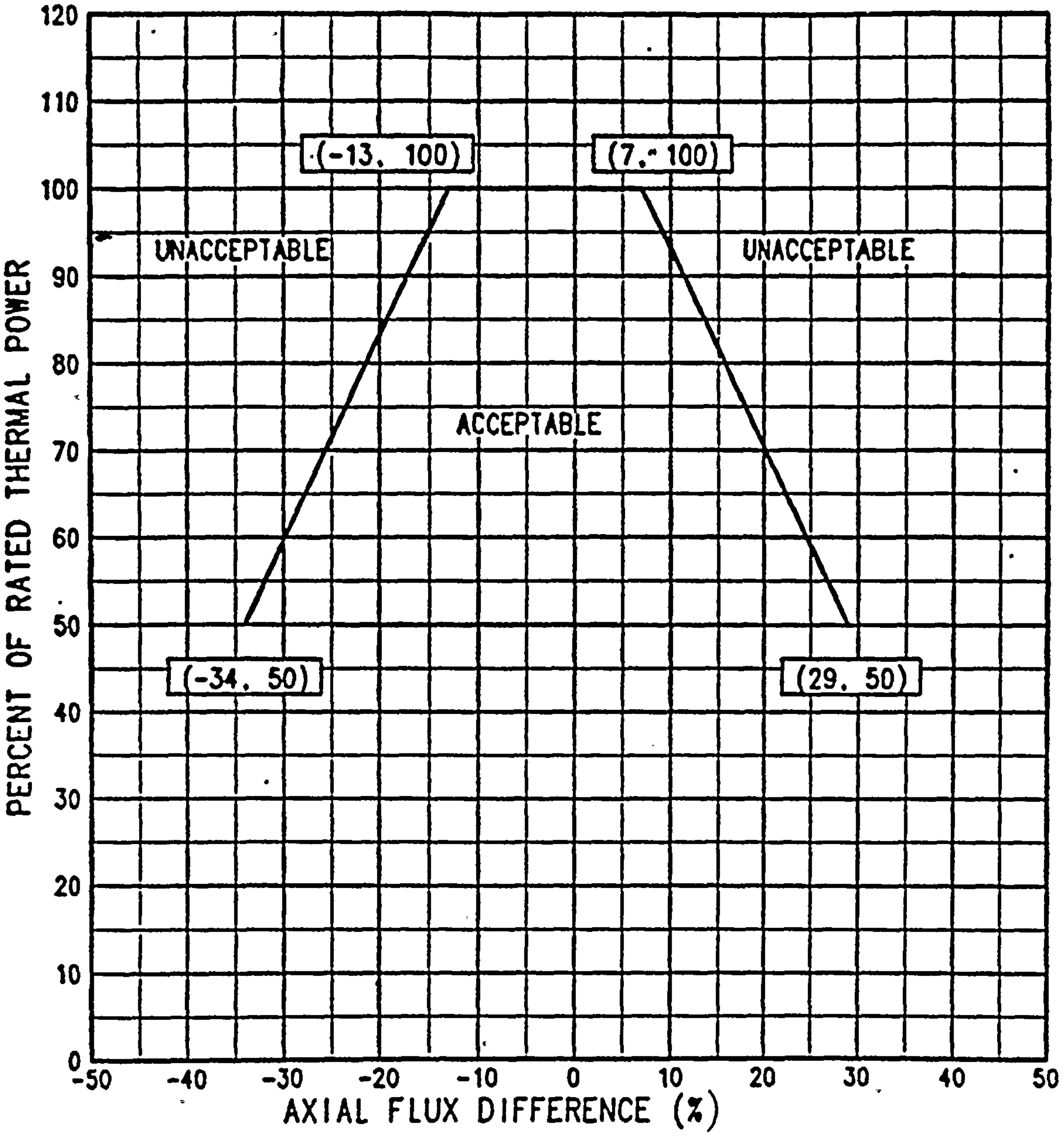

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER FOR RAOC

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z) \le \frac{\overset{2.32}{\cancel{2.28}}}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \le (\overset{4.64}{\cancel{4.56}}) [K(Z)] \text{ for } P \le 0.5$$

Where:

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

K(Z) = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

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

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

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{2.32}{P \times W(Z)} \times K(Z) \text{ for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{2.32}{W(Z) \times 0.5} \times K(Z) \text{ for } P \leq 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.32 is the F_Q limit, K(Z) is given in Figure 3.2-2, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6.

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

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

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

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

e. With measurements indicating

$$\text{maximum} \left(\frac{F_Q^M(Z)}{K(Z)}\right)$$

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

1) $F_Q^M(Z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c. or

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

$$\text{maximum} \left(\frac{F_Q^M(Z)}{K(Z)}\right) \text{ is not increasing.}$$

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

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

$$\left\{\left(\text{maximum} \left[\frac{F_Q^M(Z) \times W(Z)}{\frac{2.32}{P} \times K(Z)}\right]\right) - 1\right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{\left(\text{maximum} \left[\frac{F_Q^M(Z) \times W(Z)}{\frac{2.32}{0.5} \times K(Z)}\right]\right) - 1\right\} \times 100 \text{ for } P < 0.5$$

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits of 3.2-1 by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

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

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

SURVEILLANCE REQUIREMENTS (Continued)

g. The limits specified in Specifications 4.2.2.2c, 4.2.2.2e, and 4.2.2.2f above are not applicable in the following core plane regions:

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

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

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within $\pm$ 3% of target flux difference) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \text{minimum}\left[\frac{2.32 \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}}\right] \times 100\%$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is 2.32. K(Z) is given in Figure 3.2-2. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. The function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6.

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

4.2.2.4 During Base Load operation $F_Q(Z)$ shall be evaluated to determine if it is within its limit by:

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

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

SURVEILLANCE REQUIREMENTS (Continued)

c. Satisfying the following relationship:

$$F_Q^M(Z) \le \frac{2.32 \times K(Z)}{P \times W(Z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. The F_Q limit is 2.32.

K(Z) is given in Figure 3.2-2. P is the fraction of RATED THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6.

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

1. Prior to entering Base Load operation after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and

2. At least once per 31 effective full power days.

e. With measurements indicating

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

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

$$[(\text{max. of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{2.32}{P} \times K(Z)}]) - 1] \times 100 \text{ for } P \geq APL^{ND}$$

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

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

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

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

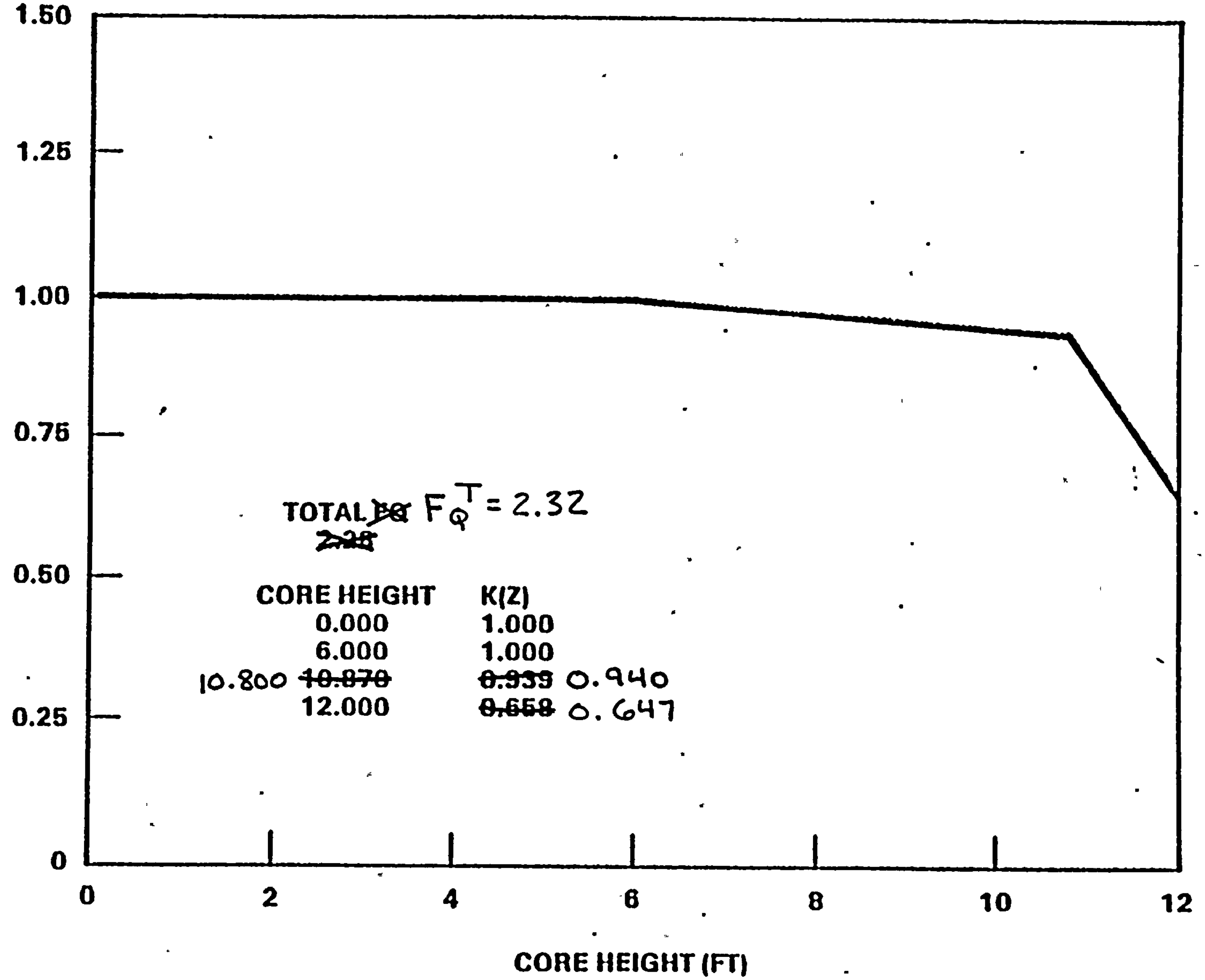

FIGURE 3.2-2

K(Z) - LOCAL AXIAL PENALTY FUNCTION FOR $F_Q(Z)$

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

LIMITING CONDITION FOR OPERATION

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

a. Measured RCS flow rate $\geq$ 292,800 gpm x $(1.0 + C_1)$, and

b. Measured $F^N_{\Delta H} \leq 1.49\ [1.0 + \cancel{0.2}(1.0-P)]$
0.3

Where:

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

$F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map, and the measured values of $F^N_{\Delta H}$ shall be used for comparison above since the 1.49 value above accounts for a 4% allowance on incore measurement of $F^N_{\Delta H}$.

C_1 = Measurement uncertainty for core flow as described in the Bases.

APPLICABILITY: MODE 1.

ACTION:

With RCS total flow rate or $F^N_{\Delta H}$ outside the above limits:

a. Within 2 hours either:

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

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

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(12)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), # ~~M~~(3, 4), ## ~~Q~~(4, 6), R(4, 5)	Q(15)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q(15)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q(15)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1), Q(8, 15)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	S	R(11)	Q(15)	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q(15)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(15)	N.A.	N.A.	1 (16)
10. Pressurizer Pressure--High	S	R	Q(15)	N.A.	N.A.	1, 2

Each 31 Effective Full Power Days
Each 92 Effective Full Power Days

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

***Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(9) Setpoint verification is not applicable.

(10) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the reactor trip breakers.

(11) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

(12) Verify that appropriate signals reach the undervoltage and shunt trip relays, for both the main and bypass breakers, from the manual reactor trip switch.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

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

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

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

~~$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.~~

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of ~~2.28~~ 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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

TARGET AFD

AXIAL FLUX DIFFERENCE (Continued)   REPLACE WITH ATTACHED INSERT

~~Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.~~

~~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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.~~

~~Figure B 3/4 2-1 shows a typical monthly target band.~~

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

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

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

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

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

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At power levels below APL^{ND}, the limits on AFD are defined by Figure 3.2-1, i.e. that defined by the RAOC operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g. load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible; 1) RAOC, the AFD limits of which are defined by Figure 3.2-1, and 2) Base Load operation, which is defined as the maintenance of the AFD within a $\pm 3\%$ band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(Z)$ less than its limiting value. To allow operation at the maximum permissible value, the Base Load operating procedure restricts the indicated AFD to a relatively small target band and power swings (AFD target band of $\pm 3\%$, $APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24 hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are: 1) outside the allowed ΔI power operating space (for RAOC operation), or 2) outside the acceptable AFD target band (for Base Load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) APL^{ND} (for Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

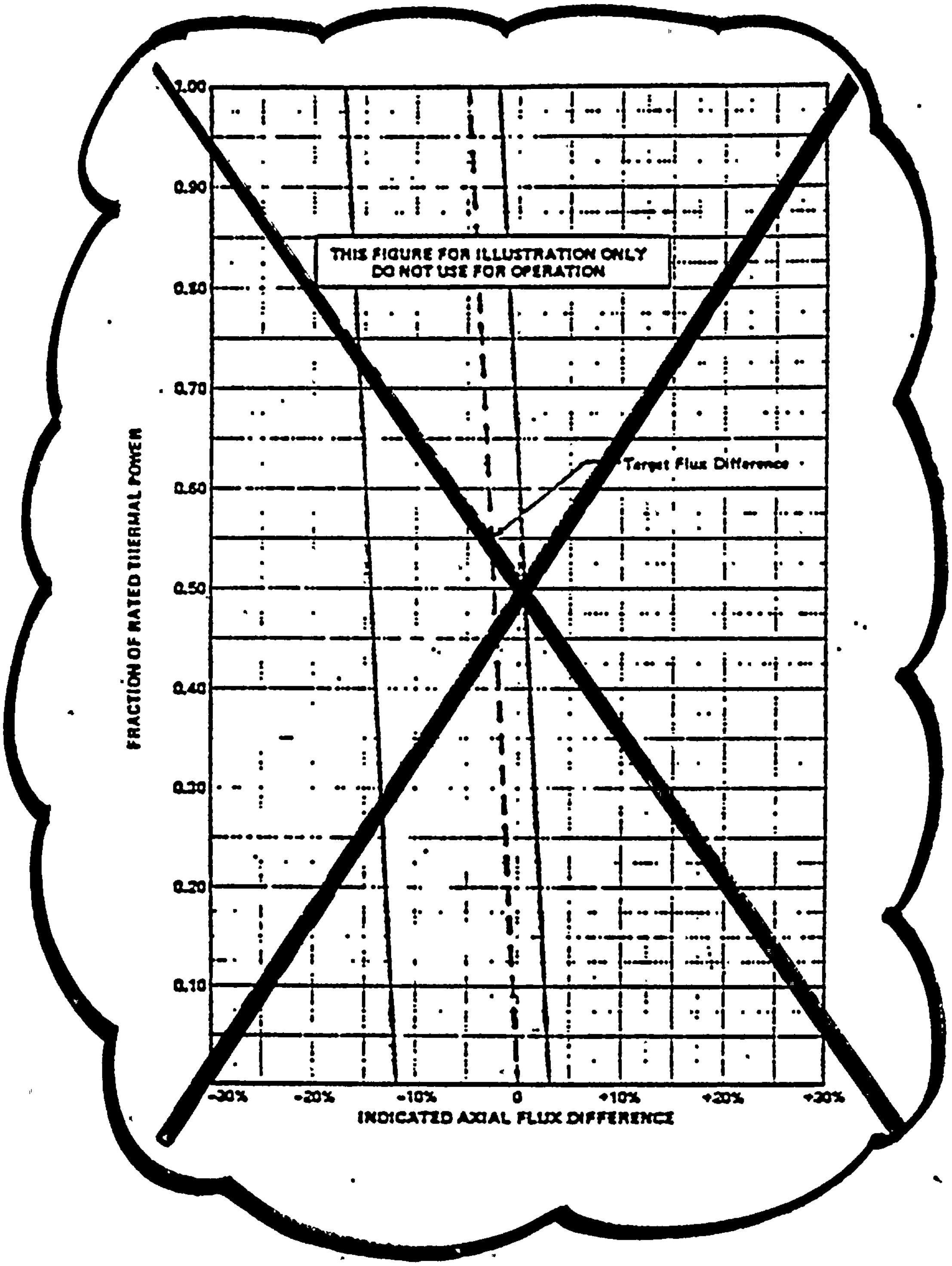

FIGURE B 3/4 2-1 (DELETED)

~~TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER FOR BURNUP GREATER THAN 6000 MWD/MTU~~

The applicable value of rod bow and any other penalties is presented in FSAR Section 4.4.2.2.4.2.

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

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

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

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The combination of the RCS flow requirement and the measurement of $F^N_{\Delta H}$ ensures that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

$F^N_{\Delta H}$ is evaluated as being less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. ~~(less than 3% for the worst case which occurs at a burnup of 33,000 MWD/MTU).~~ This margin includes the following:

a. Design limit DNBR of 1.30 vs 1.28,
b. Grid Spacing (K_s) of 0.046 vs 0.059,
c. Thermal Diffusion Coefficient of 0.038 vs 0.059,
d. DNBR Multiplier of 0.86 vs 0.88, and
e. Pitch reduction.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

~~The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F^{RTP}_{xy}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.~~

INSERT ATTACHED

INSERT FOR PAGE B 3/4 2-4

The hot channel factor $F_Q^M(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, $W(Z)$ or $W(Z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(Z)$, is met. $W(Z)$ accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(Z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. The $W(Z)$ function for normal operation is provided in the Peaking Factor Limit Report per Specification 6.9.1.6.

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except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, stainless steel, or by vacancies may be made if justified by a cycle specific evaluation.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly Normally containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.9 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 9410 ± 100 cubic feet at a nominal T_{avg} of 588.8°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological station shall be located as shown on Figure 5.1-1.

~~RADIAL~~ PEAKING FACTOR LIMIT REPORT REPLACE WITH ATTACHED INSERT

~~6.9.1.6 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the NRC Regional Administrator with a copy to the Director of Nuclear Reactor Regulation, Attention: Chief, Reactor Systems Branch, Division of PWR Licensing-A, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, for all core planes containing Bank "D" control rods and all unrodded core planes and the plot of predicted ($F_q^T \cdot P_{Rel}$) vs Axial Core Height with the limit envelope at least 60 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter. In addition, in the event that the limit should change requiring a new substantial or an amended submittal to the Radial Peaking Factor Limit Report, it will be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.~~

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

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The W(Z) Functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be established for each reload core and implemented prior to use.

The methodology used to generate the W(Z) functions for RAOC and Base Load operation and the value for APL^{ND} shall be those previously reviewed and approved by the NRC.* If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the W(Z) functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be provided to the NRC in accordance with 10 CFR 50.4 within 30 days after each cycle initial criticality.

Any information needed to support W(Z), $W(Z)_{BL}$, and APL^{ND} will be by request from the NRC and need not be included in this report.

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* WCAP-10216 "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification".

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