INSERT 1

REPLACE WITH ATTACHED NEW FIGURE 2.1-1

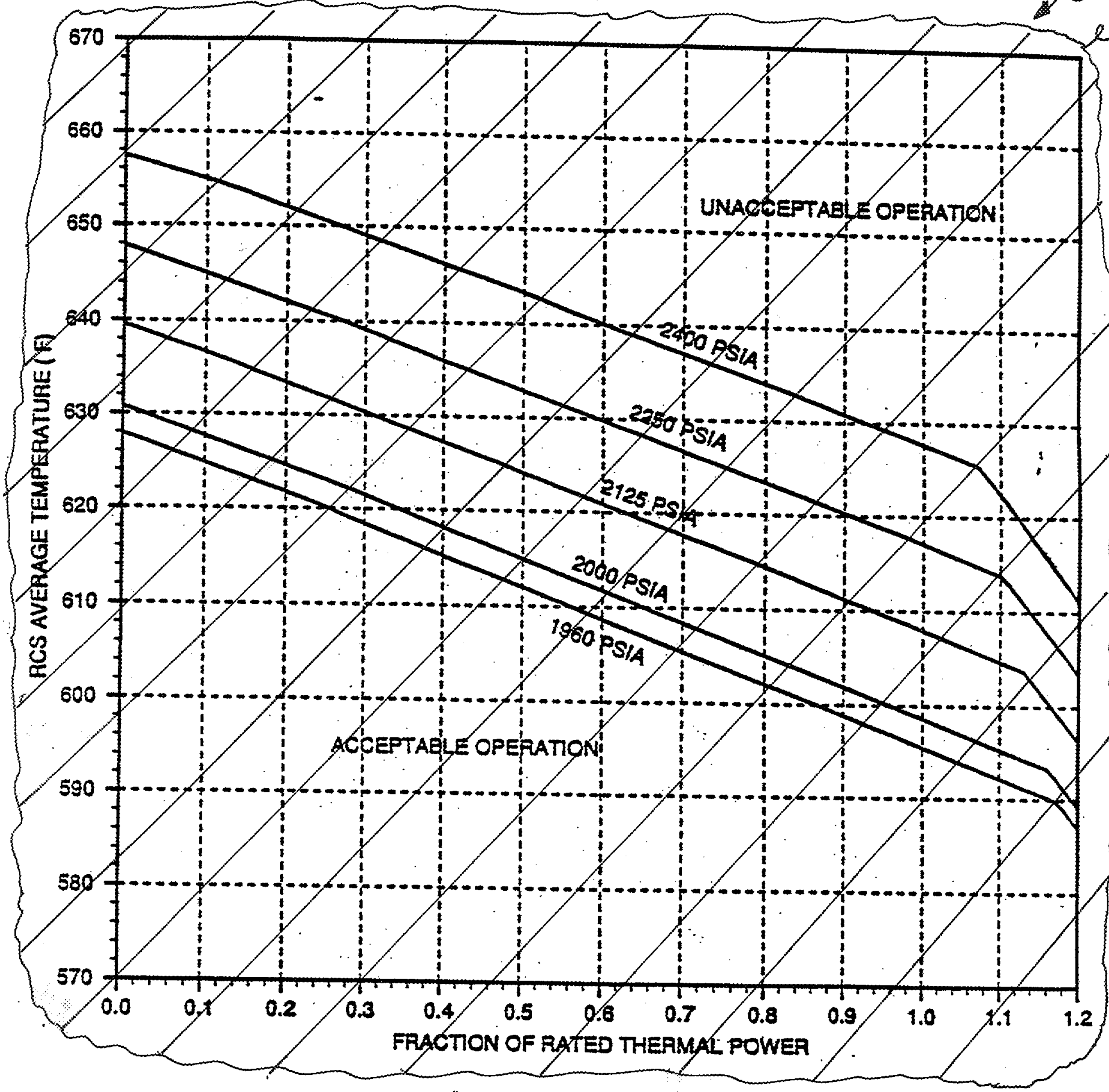

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

NEW FIGURE 2.1-1

INSERT 1

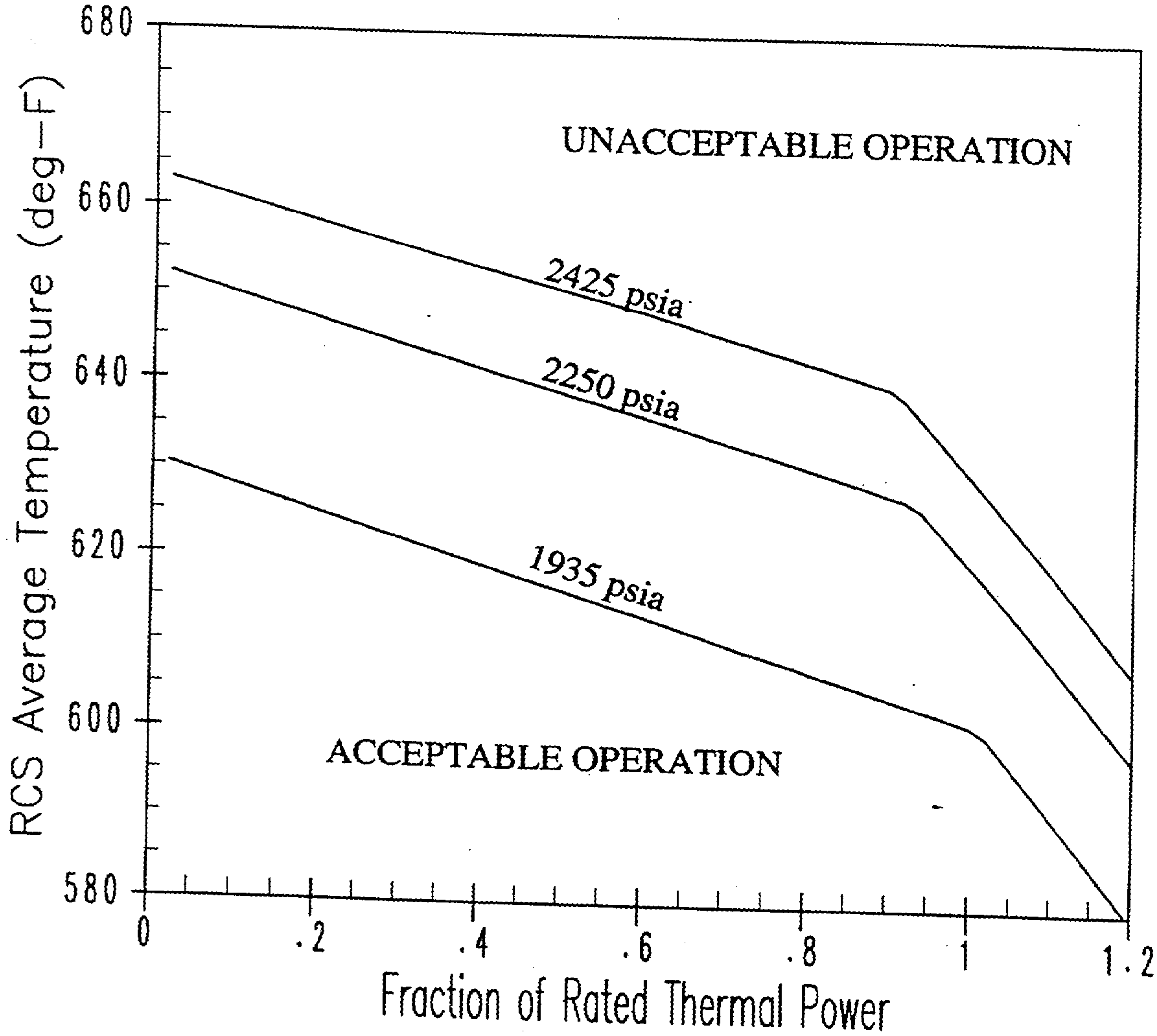

Figure 2.1-1

REACTOR CORE SAFETY LIMITS - FOUR LOOPS IN OPERATION

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

T^1 ≤ ~~588.5°F (Nominal T_{avg} at RATED THERMAL POWER)~~;

INDICATED T_{avg} AT RATED THERMAL POWER (CALIBRATION TEMPERATURE FOR ΔT INSTRUMENTATION, ≤ 588.5 °F)

K_3 = Value specified in COLR;

P = Pressurizer pressure, psig;

P^1 = 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 as specified in the COLR.

NOTE 2: Cycle dependent values for the channel's Allowable Value are specified in the COLR.

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 that would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

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

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

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

VALUES OF THE

These curves are based on ~~an~~ enthalpy rise hot channel factor $F^N_{\Delta H}$, at RATED THERMAL POWER, ~~of 1.65~~. The value of $F^N_{\Delta H}$ at reduced power is assumed to vary according to the expression: FOR THE VALUES SPECIFIED IN THE COLR.

$F^N_{\Delta H}(RTP)$

$$F^N_{\Delta H} = \cancel{1.65}\,[1 + 0.3\,(1-P)]$$

~~Where~~ P is the fraction of RATED THERMAL POWER.

This expression conservatively bounds the cycle specific limits on $F^N_{\Delta H}$ specified in Technical Specification 3/4.2.3 and the COLR. The Safety Limits in Figure 2.1-1 are also based on a reference cosine axial power shape with a peak of 1.55.

WHERE:
$F^N_{\Delta H}(RTP)$ IS THE VALUE AT RATED THERMAL POWER AND

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power, a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than ~~1.30~~ OR EQUAL TO THE DNBR LIMITS SPECIFIED IN THE APPLICABLE NRC-APPROVED ANALYTICAL METHODS REFERENCED IN SPECIFICATION 6.8.1.6.b.

3A7 91 48

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 limits specified in the COLR~~, with the Fixed Incore Detector (FIDS) Alarm OPERABLE, or~~.

~~b. The limits specified in the COLR, when the FIDS Alarm is inoperable.~~

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER.

ACTION:

a. With the indicated AFD* outside of the applicable limits specified in the COLR:

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

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

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

~~b. With an OPERABLE FIDS Alarm exceeding in limit:~~

~~1. Comply with the AFD limits specified in the COLR for operation with the FIDS Alarm inoperable within 15 minutes and,~~

~~2. Verify THERMAL POWER is less than the maximum power limit established by Surveillance Requirement 4.2.1.2 within 15 minutes and,~~

~~3. Identify and correct the cause of the FIDS Alarm prior to operation beyond the limits specified in the COLR for operation with the FIDS Alarm inoperable.~~

~~c. With the FIDS Alarm inoperable, within 4 hours,~~

~~1. Comply with the AFD limits specified in the COLR for operation with the FIDS Alarm inoperable, and~~

~~2. Verify THERMAL POWER is less than the maximum power limit established by Surveillance Requirement 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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

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 at least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

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 At least once per 31 EFPD determine the maximum allowed power for operation with the FIDS Alarm inoperable by comparing $F_Q(Z)$ to the $F_Q(Z)$ limit established for operation with the FIDS Alarm inoperable.~~

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

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

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

$$F_Q(Z) \le \frac{F_Q^{RTP}}{.5} K(Z) \text{ for } P \le 0.5$$

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

F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the COLR, and

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

APPLICABILITY: MODE 1.

ACTION:

a. With $F_Q(Z)$ exceeding its limit:

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

2. ~~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

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

~~LIMITING CONDITION FOR OPERATION~~ SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 ~~$F_Q(Z)$ shall be demonstrated to be within its limits prior to operation above 75% RATED THERMAL POWER after each fuel loading and at least once per 31 EFPD thereafter by:~~

INSERT 2

~~a. Using the Incore Detector System 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% when using the movable incore detectors or 5.21% when using the fixed incore detectors, to account for measurement uncertainties.~~

4.2.2.3 ~~The limits of Specification 3.2.2 are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:~~

INSERT 2A

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

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

4.2.2.4 When being used, Each fixed incore detector alarm setpoint shall be updated at least once per 31 EFPD. The alarm setpoints will be based on the latest available power distribution, so that the alarm setpoint does not exceed the $F_Q(Z)$ limit defined in Technical Specification 3.2.2.

INSERT (2)

4.2.2.2 $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limits by:

a. Using the 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% when using the moveable incore detectors or 5.21% when using the fixed incore detectors to account for measurement uncertainties.

c. Satisfying the following relationship:

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

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

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

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

1. Upon achieving equilibrium conditions after exceeding by 20% 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 (EFPD), whichever occurs first.

NOTE: MAKE ALL (Z) UPPERCASE

*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.

INSERT (2) CONTINUED

e. With measurements indicating that the maximum over the elevation z of $\frac{F_Q^M(z)}{K(z)}$ has increased since the previous determination of $F_Q^M(z)$ one of the following actions shall be taken:

1) Continuously monitor the Fixed Incore Detector (FIDs) Alarm, when the alarm is OPERABLE, or

2) Increase $F_Q^M(z)$ by the appropriate factor specified in the COLR prior to confirming the relationship specified in Specification 4.2.2.2.c, or

3) $F_Q^M(z)$ shall be measured at least once per 7 EFPD until two successive maps indicate that the maximum over the elevation z of $\frac{F_Q^M(z)}{K(z)}$ is not increasing.

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

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

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

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

2) Place the core in an equilibrium condition where the limit in Specification 4.2.2.2.c is satisfied within 2 hours. Power level may then be increased provided the AFD limits of Specification 3.2.1 are reduced 1% AFD for each percent $F_Q(z)$ exceeds it limit.

INSERT 2 CONTINUED

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

INSERT 2A

4.2.2.3 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% when using the moveable incore detectors or 5.21% when using the fixed incore detectors to account for measurement uncertainty.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be less than the limits specified in the COL~~A~~R.

OR EQUAL TO

APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours reduce the THERMAL POWER to the level where the LIMITING CONDITION FOR OPERATION is satisfied.

b. ~~Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the limit required by ACTION a., above;~~ THERMAL POWER may ~~then~~ be increased, provided $F^N_{\Delta H}$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be demonstrated to be within its limit prior to operation above 75% RATED THERMAL POWER after each fuel loading and at least once per 31 EFPD thereafter by:

a. Using the Incore Detector System to obtain a power distribution map at any THERMAL POWER greater than 5% RATED THERMAL POWER.

b. Using the measured value of $F^N_{\Delta H}$ which does not include an allowance for measurement uncertainty.

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

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The design limit DNBR includes margin to offset any rod bow penalty. Margin is also maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is available for plant design flexibility.

When an F_Q measurement is taken, an allowance for both measurement error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the movable incore detectors, while 5.21% is appropriate for surveillance results determined with the fixed incore detectors. A 3% allowance is appropriate for manufacturing tolerance.

~~For operation with the Fixed Incore Detector System (FIDS) Alarm OPERABLE, the cycle-dependent normalized axial peaking factor, K(Z), specified in COLR accounts for axial power shape sensitivity in the LOCA analysis. Assurance that the $F_Q(Z)$ limit on Specification 3.2.2 is met during both normal operation and in the event of xenon redistribution following power changes is provided by the FIDS Alarm through the plant process computer. This assures that the consequences of a LOCA would be within specified acceptance criteria.~~

~~For operation with the FIDS Alarm inoperable, the cycle-dependent normalized axial peaking factor, K(Z), specified in COLR accounts for possible xenon redistribution following power changes in addition to axial power shape sensitivity in the LOCA analysis. This assures that the consequences of a LOCA would be within specified acceptance criteria.~~

INSERT 3

When RCS $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the established limit. A bounding measurement error of 4.13% for $F^N_{\Delta H}$ has been allowed for in determination of the design DNBR value.

3/4.2.4 QUADRANT POWER TILT RATIO

The purpose of this specification is to detect gross changes in core power distribution between monthly Incore Detector System surveillances. During normal operation the QUADRANT POWER TILT RATIO is set equal to zero once acceptability of core peaking factors has been established by review of incore surveillances. The limit of 1.02 is established as an indication that the power distribution has changed enough to warrant further investigation.

INSERT 3

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to (RAOC) operation, W(z), 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. The W(z) function for normal operation is specified in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6.

RELAXED AXIAL OFFSET CONTROL

6.8.1.6.a. (Continued)

5. Shutdown Rod Insertion limit for Specification 3.1.3.5,
6. Control Rod Bank Insertion limits for Specification 3.1.3.6,
7. AXIAL FLUX DIFFERENCE limits for Specification 3.2.1,
8. Heat Flux Hot Channel Factor, F^{RTP}_{Q} and K(Z) for Specification 3.2.2,
9. Nuclear Enthalpy Rise Hot Channel Factor, and $F^{RTP}_{\Delta H}$ for Specification 3.2.3.

The CORE OPERATING LIMITS REPORT shall be maintained available in the Control Room.

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

1. WCAP-10266-P-A, Rev. 2 with Addenda (Proprietary) and WCAP-11524-A (Nonproprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", August, 1986

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

2. WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Nonproprietary), "NOTRUMP: A Nodal Transient Small Break and General Network Code", August, 1985

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

3. YAEC-1363-A, "CASMO-3G Validation," April, 1988.

   YAEC-1659-A, "SIMULATE-3 Validation and Verification," September, 1988.

   INSERT 4

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

4. Seabrook Station Updated Final Safety Analysis Report, Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant System".

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5

6.8.1.6.b. (Continued)

INSERT (5)

5. YAEC-1241, "Thermal-Hydraulic Analysis of PWR Fuel Elements Using the CHIC-KIN Code", R. E. Helfrich, March, 1981

   Methodology for Specification:
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

INSERT (6)

6. YAEC-1849P, "Thermal-Hydraulic Analysis Methodology Using VIPRE-01 For PWR Applications, "October, 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

7. YAEC-1854P, "Core Thermal Limit Protection Function Setpoint Methodology For Seabrook Station, "October, 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

8. YAEC-1856P, "System Transient Analysis Methodology Using RETRAN for PWR Applications," December, 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

9. YAEC-1752, "STAR Methodology Application for PWRs, Control Rod Ejection, Main Steam Line Break," October, 1990

   Methodology for Specification:
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.b. (Continued)

10. YAEC-1855P, "Seabrook Station Unit 1 Fixed Incore Detector System Analysis," October, 1992

Methodology for Specification:
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

11. YAEC-1624P, "Maine Yankee RPS Setpoint Methodology Using Statistical Combination of Uncertainties - Volume 1 - Prevention of Fuel Centerline Melt," March, 1988

INSERT 7

Methodology for Specification:
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

12. NYN-95048, Letter from T. C. Feigenbaum (NAESCo) to NRC, "License Amendment Request 95-05: Positive Moderator Temperature Coefficient", May 30, 1995

Methodology for Specification:
3.1.1.3- Moderator Temperature Coefficient

13. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report", April, 1995, (Westinghouse Proprietary)

Methodology for Specification:
3.2.2- Heat Flux Hot Channel Factor

INSERT 8

6.8.1.6.c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT for each reload cycle, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector.

NOTE: THESE INSERTS ARE TO BE ADDED TO ITS RESPECTIVE ITEM NUMBER

INSERT (4) →

3. WCAP-11596-P-A, (Proprietary), "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores", June, 1988

WCAP-10965-P-A, (Proprietary), "ANC: A Westinghouse Advanced Nodal Computer Code", September, 1986

INSERT (5) →

5. WCAP-14565-P, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", April, 1997

Letter from T. H. Essig (NRC) to H. Sepp (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-14565-P, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", January, 1999

INSERT (6) →

6. WCAP-11397-P-A, (Proprietary), "Revised Thermal Design Procedure", April, 1989

INSERT (7) →

11. WCAP-14551-P, (Proprietary), "Westinghouse Setpoint Methodology for Protection Systems, Seabrook Nuclear Power Station Unit 1, 24 Month Fuel Cycle Evaluation", June, 1998

INSERT (8) →

14. WCAP-10216-P-A, Revision 1A (Proprietary), "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification", February, 1994

WCAP-8385-P, (Proprietary), "Power Distribution Control and Load Following Procedures", September, 1974

Methodology for Specifications:
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor

15. WCAP-9272-P-A, (Proprietary), "Westinghouse Reload Safety Evaluation Methodology", July, 1985

Methodology for Specifications:
3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
3.1.1.2 - SHUTDOWN MARGIN for MODE 5
3.1.1.3 - Moderator Temperature Coefficient
3.1.3.5 - Shutdown Rod Insertion Limit
3.1.3.6 - Control Rod Insertion Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

SECTION III

Retype of the Proposed Change

The attached retype reflects the currently issued version of the Technical Specifications. Pending Technical Specification changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with the Technical Specifications prior to issuance.

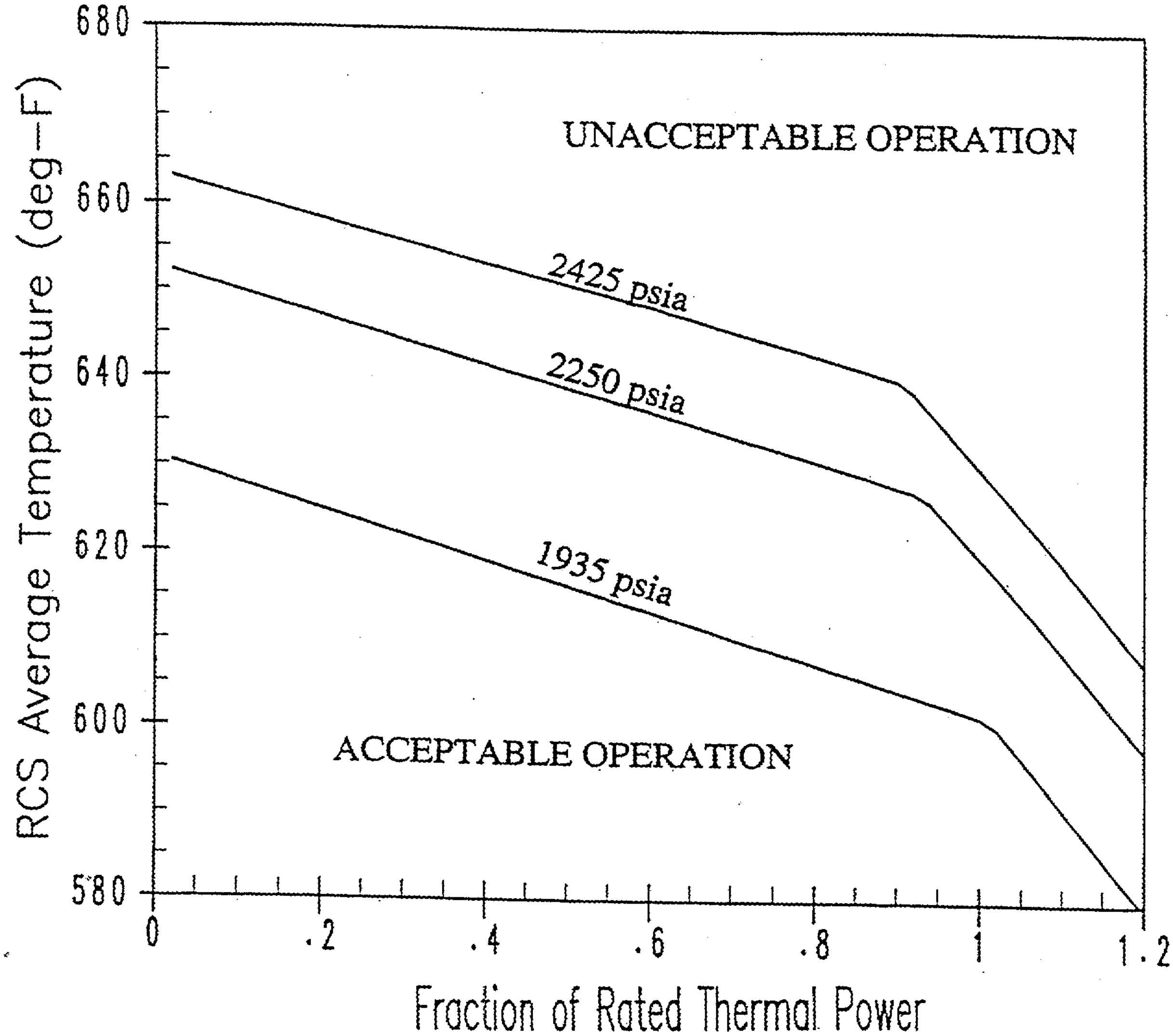

Figure 2.1-1

REACTOR CORE SAFETY LIMITS—FOUR LOOPS IN OPERATION

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

T^1	$\leq$	Indicated T_{avg} at RATED THERMAL POWER; (Calibration Temperature for ΔT Instrumentation, $\leq$ 588.5°F);
K_3	=	Value specified in COLR;
P	=	Pressurizer pressure, psig;
P^1	=	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 as specified in the COLR.

NOTE 2: Cycle dependent values for the channel's Allowable Value are specified in the COLR.

2.1.1 REACTOR CORE

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

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

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

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

These curves are based on values of the enthalpy rise hot channel factor $F^N_{\Delta H}$, at RATED THERMAL POWER, for the values specified in the COLR. The value of $F^N_{\Delta H}$ at reduced power is assumed to vary according to the expression:

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

Where:

$F^N_{\Delta H}$ (RTP) is the value at RATED THERMAL POWER, and
P is the fraction of RATED THERMAL POWER.

This expression conservatively bounds the cycle specific limits on $F^N_{\Delta H}$ specified in Technical Specification 3/4.2.3 and the COLR. The Safety Limits in Figure 2.1-1 are also based on a reference cosine axial power shape with a peak of 1.55.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power, a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than or equal to the DNBR limits specified in the applicable NRC-approved analytical methods referenced in Specification 6.8.1.6.b.

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 the limits specified in the COLR.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER.

ACTION:

a. With the indicated AFD* outside of the applicable limits specified in the COLR:

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

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

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

*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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

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 at least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

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.

3/4 2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

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

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

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{.5} K(Z) \text{ for } P \leq 0.5$$

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

F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the COLR, and

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

APPLICABILITY: MODE 1.

ACTION:

a. With $F_Q(Z)$ exceeding its limit:

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

2. THERMAL power may 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 $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits by:

a. Using the 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% when using the moveable incore detectors or 5.21% when using the fixed incore detectors to account for measurement uncertainties.

c. Satisfying the following relationship:

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

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

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

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

1) Upon achieving equilibrium conditions after exceeding by 20% 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 (EFPD), whichever occurs first.

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

SURVEILLANCE REQUIREMENTS

e. With measurements indicating that the maximum over the elevation Z of $\frac{F_Q^M(Z)}{K(Z)}$ has increased since the previous determination of $F_Q^M(Z)$ one of the following actions shall be taken:

1) Continuously monitor the Fixed Incore Detector (FIDs) Alarm, when the alarm is OPERABLE, or

2) Increase $F_Q^M(Z)$ by the appropriate factor specified in the COLR prior to confirming the relationship specified in Specification 4.2.2.2.c, or

3) $F_Q^M(Z)$ shall be measured at least once per 7 EFPD until two successive maps indicate that the maximum over the elevation Z of $\frac{F_Q^M(Z)}{K(Z)}$ is not increasing.

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

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

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

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

2) Place the core in an equilibrium condition where the limit in Specification 4.2.2.2.c is satisfied within 2 hours. Power level may then be increased provided the AFD limits of Specification 3.2.1 are reduced 1% AFD for each percent $F_Q(Z)$ exceeds it limit.

SURVEILLANCE REQUIREMENTS

g. The limits specified in Specification 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f 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 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% when using the moveable incore detectors or 5.21% when using the fixed incore detectors to account for measurement uncertainty.

4.2.2.4 When being used, each fixed incore detector alarm setpoint shall be updated at least once per 31 EFPD. The alarm setpoints will be based on the latest available power distribution, so that the alarm setpoint does not exceed the $F_Q(Z)$ limit defined in Technical Specification 3.2.2.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours reduce the THERMAL POWER to the level where the LIMITING CONDITION FOR OPERATION is satisfied.

b. THERMAL POWER may be increased, provided $F^N_{\Delta H}$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be demonstrated to be within its limit prior to operation above 75% RATED THERMAL POWER after each fuel loading and at least once per 31 EFPD thereafter by:

a. Using the Incore Detector System to obtain a power distribution map at any THERMAL POWER greater than 5% RATED THERMAL POWER.

b. Using the measured value of $F^N_{\Delta H}$ which does not include an allowance for measurement uncertainty.

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

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The design limit DNBR includes margin to offset any rod bow penalty. Margin is also maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is available for plant design flexibility.

When an $F_Q(Z)$ measurement is taken, an allowance for both measurement error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the movable incore detectors, while 5.21% is appropriate for surveillance results determined with the fixed incore detectors. A 3% allowance is appropriate for manufacturing tolerance.

The hot channel factor $F^M_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to Relaxed Axial Offset Control (RAOC) operation, W(Z), 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. The W(Z) function for normal operation is specified in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6.

When RCS $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the established limit. A bounding measurement error of 4.13% for $F^N_{\Delta H}$ has been allowed for in determination of the design DNBR value.

3/4.2.4 QUADRANT POWER TILT RATIO

The purpose of this specification is to detect gross changes in core power distribution between monthly Incore Detector System surveillances. During normal operation the QUADRANT POWER TILT RATIO is set equal to zero once acceptability of core peaking factors has been established by review of incore surveillances. The limit of 1.02 is established as an indication that the power distribution has changed enough to warrant further investigation.

6.8.1.6.a. (Continued)

5. Shutdown Rod Insertion limit for Specification 3.1.3.5,

6. Control Rod Bank Insertion limits for Specification 3.1.3.6,

7. AXIAL FLUX DIFFERENCE limits for Specification 3.2.1,

8. Heat Flux Hot Channel Factor, F_Q^{RTP} and K(Z) for Specification 3.2.2,

9. Nuclear Enthalpy Rise Hot Channel Factor, and $F^{RTP}_{\Delta H}$ for Specification 3.2.3.

The CORE OPERATING LIMITS REPORT shall be maintained available in the Control Room.

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

1. WCAP-10266-P-A, Rev. 2 with Addenda (Proprietary) and WCAP-11524-A (Nonproprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", August, 1986

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

2. WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Nonproprietary), "NOTRUMP: A Nodal Transient Small Break and General Network Code", August, 1985

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

3. YAEC-1363-A, "CASMO-3G Validation," April,1988.

   YAEC-1659-A, "SIMULATE-3 Validation and Verification," September, 1988.

   WCAP-11596-P-A (Proprietary), "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores", June, 1988.

   WCAP-10965-P-A (Proprietary), "ANC: A Westinghouse Advanced Nodal Computer Code", September, 1986.

6.8.1.6.b. (Continued)

Methodology for Specifications:
3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
3.1.1.2 - SHUTDOWN MARGIN for MODE 5
3.1.1.3 - Moderator Temperature Coefficient
3.1.3.5 - Shutdown Rod Insertion Limit
3.1.3.6 - Control Rod Insertion Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

4. Seabrook Station Updated Final Safety Analysis Report, Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant System".

Methodology for Specifications:
3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
3.1.1.2 - SHUTDOWN MARGIN for MODE 5

5. YAEC-1241, "Thermal-Hydraulic Analysis of PWR Fuel Elements Using the CHIC-KIN Code", R. E. Helfrich, March, 1981

WCAP-14565-P, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", April, 1997

Letter from T. H. Essig (NRC) to H. Sepp (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-14565-P, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", January, 1999

Methodology for Specification:
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6. YAEC-1849P, "Thermal-Hydraulic Analysis Methodology Using VIPRE-01 For PWR Applications," October, 1992

WCAP-11397-P-A, (Proprietary), "Revised Thermal Design Procedure", April, 1989

6.8.1.6.b. (Continued)

Methodology for Specification:
2.2.1 - Limiting Safety System Settings
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

7. YAEC-1854P, "Core Thermal Limit Protection Function Setpoint Methodology For Seabrook Station," October, 1992

Methodology for Specification:
2.2.1 - Limiting Safety System Settings
3.1.3.5 - Shutdown Rod Insertion Limit
3.1.3.6 - Control Rod Insertion Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

8. YAEC-1856P, "System Transient Analysis Methodology Using RETRAN for PWR Applications," December, 1992

Methodology for Specification:
2.2.1 - Limiting Safety System Settings
3.1.1.3 - Moderator Temperature Coefficient
3.1.3.5 - Shutdown Rod Insertion Limit
3.1.3.6 - Control Rod Insertion Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

9. YAEC-1752, "STAR Methodology Application for PWRs, Control Rod Ejection, Main Steam Line Break," October, 1990

Methodology for Specification:
3.1.1.3 - Moderator Temperature Coefficient
3.1.3.5 - Shutdown Rod Insertion Limit
3.1.3.6 - Control Rod Insertion Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.b. (Continued)

10. YAEC-1855P, "Seabrook Station Unit 1 Fixed Incore Detector System Analysis," October, 1992

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

11. YAEC-1624P, "Maine Yankee RPS Setpoint Methodology Using Statistical Combination of Uncertainties - Volume 1 - Prevention of Fuel Centerline Melt," March, 1988

    WCAP-14551-P, (Proprietary), "Westinghouse Setpoint Methodology for Protection Systems, Seabrook Nuclear Power Station Unit 1, 24 Month Fuel Cycle Evaluation", June, 1998

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

12. NYN-95048, Letter from T. C. Feigenbaum (NAESCo) to NRC, "License Amendment Request 95-05: Positive Moderator Temperature Coefficient", May 30, 1995

    Methodology for Specification:
    3.1.1.3 - Moderator Temperature Coefficient

13. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report". April, 1995. (Westinghouse Proprietary)

    Methodology for Specification:
    3.2.2 - Heat Flux Hot Channel Factor

14. WCAP-10216-P-A, Revision 1A (Proprietary), "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification", February, 1994

    WCAP-8385-P, (Proprietary), "Power Distribution Control and Load Following Procedures", September, 1974

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor

6.8.1.6.b. (Continued)

15. WCAP-9272-P-A, (Proprietary), "Westinghouse Reload Safety Evaluation Methodology", July, 1985

Methodology for Specifications:

3.1.1.1	-	SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
3.1.1.2	-	SHUTDOWN MARGIN for MODE 5
3.1.1.3	-	Moderator Temperature Coefficient
3.1.3.5	-	Shutdown Rod Insertion Limit
3.1.3.6	-	Control Rod Insertion Limits
3.2.1	-	AXIAL FLUX DIFFERENCE
3.2.2	-	Heat Flux Hot Channel Factor
3.2.3	-	Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT for each reload cycle, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector.

Section IV

Determination of Significant Hazards for Proposed Change

IV. DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGE

License Amendment Request (LAR) 99-02 propose changes to the Seabrook Station Technical Specifications (TS) to implement the Relaxed Axial Offset Control (RAOC) strategy. The RAOC TS, developed by Westinghouse, has been previously reviewed and approved by the Nuclear Regulatory Commission (NRC). In addition, current Technical Specifications allow use of the Fixed Incore Detector System (FIDS) for monitoring incore power distributions, therefore, the RAOC TS proposed for incorporation into the Seabrook Station Technical Specifications includes adjustments to allow the continued use of the FIDS, when operable, for monitoring incore power distributions and assuring compliance with the cycle-specific Limiting Conditions for Operation (LCOs) specified in the Core Operating Limits Report (COLR).

The proposed changes are in support of North Atlantic's long-term operating strategy to refuel and operate, commencing with Cycle 8, with upgraded Westinghouse fuel with Intermediate Flow Mixers (VANTAGE+ (w/ IFMs)). Use of these fuel features has been previously approved for use in Westinghouse 4-loop pressurized water reactors.

In accordance with 10 CFR 50.92, North Atlantic has reviewed the attached proposed changes and has concluded that the changes do not involve a significant hazards consideration (SHC). The basis for the conclusion that the proposed changes do not involve a SHC is as follows:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Evaluations/analyses of accidents which are potentially affected by the parameters and assumptions associated with the fuel upgrade and RAOC strategy have shown that all design standards and applicable safety criteria will continue to be met. The consideration of these changes does not result in a situation where the design, material, and construction standards that were applicable prior to the change are altered. Therefore, the proposed changes occurring with the fuel upgrade will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident.

The proposed changes associated with the fuel upgrade and RAOC strategy do not affect plant systems such that their function in the control of radiological consequences is adversely affected. The actual plant configuration, performance of systems, and initiating event mechanisms are not being changed as a result of the proposed changes. The design standards and applicable safety criteria limits will continue to be met and therefore fission barrier integrity is not challenged. The proposed changes associated with fuel upgrade and RAOC strategy have been shown not to adversely affect the response of the plant to postulated accident scenarios. The proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the UFSAR.

Therefore, for the reasons stated above, the probability or consequences of an accident previously evaluated is not significantly increased.

2. **Create the possibility of a new or different kind of accident from any accident previously evaluated.**

The possibility for a new or different type of accident from any accident previously evaluated is not created since the proposed changes associated with the fuel upgrade and RAOC strategy do not result in a change to the design basis of any plant structure, system or component. Evaluation of the effects of the fuel upgrade and RAOC strategy have shown that all design standards and applicable safety criteria continue to be met. These proposed changes therefore do not cause the initiation of any accident nor create any new failure mechanisms. Equipment important to safety will continue to operate as designed. Component integrity is not challenged. The proposed changes do not result in any event previously deemed incredible being made credible. The fuel upgrade and RAOC strategy are not expected to result in more adverse conditions and are not expected to result in any increase in the challenges to safety systems.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Involve a significant reduction in a margin of safety.**

The proposed changes will assure continued compliance within the acceptance limits previously reviewed and approved by the NRC for use of upgraded fuel features with RAOC. All of the appropriate acceptance criteria for the various analyses and evaluations will continue to be met. Therefore, the proposed changes do not involve a signification reduction in a margin of safety.

Based on the above evaluation, North Atlantic concludes that the proposed changes do not constitute a significant hazard.

Sections V & VI

Proposed Schedule for License Amendment Issuance and Effectiveness
and
Environmental Impact Assessment

V. PROPOSED SCHEDULE FOR LICENSE AMENDMENT ISSUANCE AND EFFECTIVENESS

North Atlantic requests NRC review of License Amendment Request 99-02 and issuance of a license amendment by May 1, 2000, having immediate effectiveness and implementation at commencement of Cycle 8 operation.

VI. ENVIRONMENTAL IMPACT ASSESSMENT

North Atlantic has reviewed the proposed license amendment against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, nor increase the types and amounts of effluent that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, North Atlantic concludes that the proposed changes meet the criteria delineated in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.