North
Atlantic

North Atlantic Energy Service Corporation
P.O. Box 300
Seabrook, NH 03874
(603) 474-9521

The Northeast Utilities System

April 28, 2000

Docket No. 50-443

NYN-00045

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Seabrook Station
License Amendment Request 99-02, Revision 1,
"Operation with Relaxed Axial Offset Control"

North Atlantic Energy Service Corporation (North Atlantic) has enclosed herein License Amendment Request (LAR) 99-02, Revision 1. LAR 99-02, Revision 1, is submitted pursuant to the requirements of 10CFR50.90 and 10CFR50.4.

LAR 99-02, Revision 1, supercedes the initial submittal, LAR 99-02. The enclosed LAR 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), Reference [1].

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, Reference [2], by the Nuclear Regulatory Commission (NRC). Westinghouse and Duke Engineering & Services (DE&S) jointly performed safety evaluations/analyses, using current methodologies, to confirm acceptable use of these features with RAOC for 4-loop operation for Cycle 8 operation. The safety analysis methodologies employed by Westinghouse and DE&S have been previously reviewed and approved by the Nuclear Regulatory Commission. The joint effort will also provide appropriate cycle-specific Limiting Conditions for Operation (LCOs) consistent with RAOC for use in the COLR.

Other TS changes are proposed to incorporate additional improvements to the current Power Distribution Limits Technical Specifications, resulting from review of similar TSs contained within NUREG-1431, "Standard Technical Specifications – Westinghouse Plants."

The Station Operation Review Committee and the Nuclear Safety Audit Review Committee have reviewed LAR 99-02, Revision 1.

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

[2] Davidson, S. L. (Ed.), et al., VANTAGE 5 Fuel Assembly Reference Core Report, WCAP-10444-P-A and Appendix A, September 1985; Addendum 1-A, March 1986; Addendum 2-A, April 1988.

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As discussed in the enclosed LAR Section IV, the proposed change does not involve a significant hazard consideration pursuant to 10 CFR 50.92. A copy of this letter and the enclosed LAR has been forwarded to the New Hampshire State Liaison Officer pursuant to 10 CFR 50.91(b). North Atlantic requests NRC Staff review of LAR 99-02, Revision 1, and issuance of a license amendment by October 15, 2000 (see Section V enclosed).

North Atlantic has determined that LAR 99-02, Revision 1, meets the criteria of 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement (see Section VI enclosed).

Should you have any questions regarding this letter, please contact Mr. James M. Peschel, Manager - Regulatory Programs, at (603) 773-7194.

Very truly yours,

NORTH ATLANTIC ENERGY SERVICE CORP.

W A DiProfio

William A. DiProfio
Station Director

Enclosure

cc: H. J. Miller, NRC Regional Administrator
R. M. Pulsifer, NRC Project Manager, Project Directorate 1-2
R. K. Lorson, NRC Senior Resident Inspector

Mr. Woodbury P. Fogg, P.E., Director
New Hampshire Office of Emergency Management
State Office Park South
107 Pleasant Street
Concord, NH 03301

North Atlantic

SEABROOK STATION UNIT 1

Facility Operating License NPF-86
Docket No. 50-443

License Amendment Request No. 99-02, Revision 1,
"Operation with Relaxed Axial Offset Control"

North Atlantic Energy Service Corporation pursuant to 10CFR50.90 submits License Amendment Request 99-02, Revision 1. The following information is enclosed in support of this License Amendment Request:

- Section I - Introduction and Safety Assessment for Proposed Change
- Section II - Markup of Proposed Change
- Section III - Retype of Proposed Change
- Section IV - Determination of Significant Hazards for Proposed Change
- Section V - Proposed Schedule for License Amendment Issuance and Effectiveness
- Section VI - Environmental Impact Assessment

I, William A. DiProfio, Station Director of North Atlantic Energy Service Corporation hereby affirm that the information and statements contained within License Amendment Request 99-02, Revision 1, are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed
before me this
28 day of April, 2000

Marilyn R. Sullivan
Notary Public

William A. DiProfio
Station Director

Section I

Introduction and Safety Assessment for the Proposed Change

I. INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

A. Introduction

LAR 99-02, Revision 1, 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), Reference [3].

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, Reference [2], by the Nuclear Regulatory Commission (NRC). Westinghouse and Duke Engineering & Services (DE&S) jointly performed safety evaluations/analyses, using current methodologies, to confirm acceptable use of these features with RAOC for 4-loop operation for Cycle 8 operation. The safety analysis methodologies employed by Westinghouse and DE&S have been previously reviewed and approved by the Nuclear Regulatory Commission. The joint effort will also provide appropriate cycle-specific Limiting Conditions for Operation (LCOs) consistent with RAOC for use in the COLR.

B. Safety Assessment of Proposed Changes

As stated above, use of VANTAGE+ (w/ IFMs) and the safety analysis methodologies employed by Westinghouse and DE&S have been previously approved by the NRC and acceptable use of these features for 4-loop operation has been confirmed. Safety evaluations/analyses are based on assuming Cycle 8 and subsequent transition cycles are operated within the limits of the RAOC Technical Specification developed by Westinghouse. The Westinghouse RAOC Technical Specification has been approved by the NRC, for generic application to Westinghouse PWRs, including Seabrook Station.

Associated changes to the Updated Final Safety Analysis Report (UFSAR) will be separately implemented by North Atlantic's UFSAR change process (UFCR 99-034, "Update to Westinghouse Fuel Designed with Intermediate Flow Mixers and Safety Analyses).

The proposed TS changes are shown in Section II, "Markup of the Proposed Changes." The justifications for these changes are as follows:

Revision to TS Figure 2.1-1 thermal limit lines. The existing, approved, Reference [3], thermal-hydraulic analysis of the 17x17 VANTAGE+ (w/o IFMs) fuel is based on the Revised Thermal Design Procedure (RTDP) and the WRB-1 DNB correlation. Thus, the thermal limit lines in the Technical Specifications reflect approved analysis methodology applied to the current fuel design, VANTAGE+ (w/o IFMs). The DNB analysis of Cycle 8 and subsequent transition cores containing both 17x17 VANTAGE+ (w/o IFMs) and 17x17 VANTAGE+ (w/ IFMs) fuel assemblies has been modified to incorporate the WRB-2 DNB correlation, RTDP, and VIPRE modeling as licensed by Westinghouse (References [1], [4], [5], and [6]). Therefore, the proposed changes to the thermal limit lines reflect updated and approved thermal-hydraulic analysis methodology applied to the new fuel, VANTAGE+ (w/IFMs).

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

Revisions to TS Table 2.2-1, Table Notations, clarify that specific temperature and pressure measurements are associated with the Reactor Coolant System (RCS), added the word 'measured' to "T" and "P" notations consistent with improved Standard Technical Specifications – Westinghouse Plants, NUREG-1431, Rev. 1, and relocates additional cycle-specific values for temperature, pressure and time constants to the COLR. Other cycle-specific values noted in TS Table 2.2-1 are currently located in the COLR. The revision is based on a NRC-approved Westinghouse topical report for expanding the COLR, Reference [7]. The report provides the justification to support the Technical Specification changes required to expand current COLRs to include cycle-specific RCS related Technical Specification parameter Limits. This will allow North Atlantic the flexibility to enhance plant operating margin and/or core design margins without the need for LAR submittals when making changes to cycle-specific parameters associated with the Reactor Trip System Instrumentation trip functions for the overtemperature delta-temperature (OTΔT) and overpressure delta-temperature (OPΔT).

Revision to TS 3.2.1 deleting LCOs a. & b. and ACTIONS b. & c. associated with FIDS Alarm. Currently, the COLR provides two sets of AFD limits as a function of RTP dependent on the operability status of the FIDS Alarm. The proposed change in F_Q methodology using the RAOC strategy does not provide for different AFD limits dependent on FIDS Alarm operability status, therefore, LCOs and ACTIONS associated with FIDS Alarm operability requirements are deleted.

Another revision to TS 3.2.1, deletes the requirement in ACTION a.2 requiring the reduction of the Power Range Neutron Flux – High Setpoints. The existing requirement in TS 3.2.1, Action a.2, to reduce the power range neutron flux high trip setpoints is proposed to be deleted so as to be consistent with NUREG-1431, "Standard Technical Specifications – Westinghouse Plants." Reducing the power range neutron flux high setpoint is not required to provide an adequate level of protection. Reducing the power level to less than or equal to 50 percent rated thermal power (RTP) maintains the plant in a benign condition since under RAOC methodology there are no axial flux difference (AFD) limits below 50 percent of RTP. In addition, a rapid rise in power to greater than 50 percent RTP with AFD outside limits does not immediately create an unacceptable situation. Since the transient analysis setpoint calculations for f (ΔI) (input to the overtemperature delta-temperature (OTΔT) trip function) are based on the same core power distributions that the fuel designers use for a reload cycle design, the OTΔT trip function provides an acceptable level of protection for such an excursion. It is also noted that the event would be successfully terminated by a trip at the previous setpoint level. Therefore, maintaining this provision as part of TS 3.2.1, Action a.2 is not warranted. The NRC, Reference [8], approved a similar TS change request by Southern Nuclear Operating Company for the Joseph M. Farley Nuclear Plant, dated June 12, 1996. In addition, justification of this deletion is based on Westinghouse Owners Group (WOG) letter OG-90-54 to the NRC (Jose Calvo) dated September 5, 1990.

Revision to TS 4.2.1.2 deletes the surveillance requirement (SR) for determining the maximum allowed power for operation by comparing $F_Q(Z)$ to the $F_Q(Z)$ limit established for operation with the FIDS Alarm inoperable. The proposed change in F_Q methodology using the RAOC strategy does not provide for different AFD limits dependent on FIDS Alarm operability status. The surveillance requirements associated with the proposed F_Q methodology, which is performed every 31 EFPD, bounds the surveillance requirements associated with monitoring indicated AFD regardless of FIDS Alarm operability status, therefore this SR is no longer required.

Revision to TS 3.2.2 ACTION a.2 and TS 3.2.3 ACTION b. deletes the need to identify and correct the cause of the out-of-limit condition prior to increasing Thermal Power. The explicit requirement to identify and correct the cause of the out-of-limit condition is not included in similar TSs contained in the improved Standard Technical Specifications – Westinghouse Plants, NUREG-1431, Rev. 1, and proposed

Draft Rev. 2. The basis for the deletion is that it is implicit that the out-of-limit condition would have to be corrected in order to restore compliance with the LCO. This change is considered an editorial simplification of the specifications. The NRC also approved this TS change for the Joseph M. Farley Nuclear Plant, Reference [8].

SRs 4.2.2.2 and 4.2.2.3 are totally revised to incorporate the RAOC strategy to determine $F_Q(Z)$ is within its limits. 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)). As part of this strategy, Westinghouse will be performing the supporting analysis. Westinghouse analysis methods use the Westinghouse developed RAOC strategy to determine FQ(Z) within its limits for use with these upgraded fuel features at Seabrook Station. Use of these upgraded fuel features in combination with Westinghouse developed RAOC strategy to determine $F_Q(Z)$ within its limits has been previously approved by the NRC for use at other Westinghouse 4-loop plants.

An editorial change has been made to the title, 'Limiting Condition For Operation,' on page 3/4 2-6. The title is revised to 'Surveillance Requirements,' since the following specifications are surveillance requirements and not limiting conditions.

SR 4.2.2.4 associated with updating the FIDS Alarm setpoint when the FIDS Alarm is being relied upon to extend the surveillance frequency for monitoring indicated AFD is deleted. Use of RAOC strategy does not rely on the operability of FIDS to establish operational limits for AFD.

Revision to TS 3.2.3 corrects a typo in the acronym "COLA" and corrects the upper limit on $F^N_{\Delta H}$. The changes are editorial changes. COLA is changed to COLR, the acronym for the Core Operating Limits Report, the document specifying the cycle-specific core operating limits. The other change, inclusion of $\leq$, is to make the upper limit on $F^N_{\Delta H}$ consistent with the $F^N_{\Delta H}$ value used in the safety analysis. That is, to recognize that full power operation may continue with $F^N_{\Delta H}$ equal to the upper $F^N_{\Delta H}$ limits specified in the COLR. The limits specified in the COLR are based on safety analysis limits developed using NRC-approved methodologies specified in TS 6.8.1.6.

Revision to Specification 6.8.1.6.b referencing both DE&S and Westinghouse approved reload analysis methodologies used. Specification 6.8.1.6.b lists the approved analysis methodologies used for determining the cycle specific core operating limits specified in the COLR. The current methodologies employed by DE&S are retained and the methodologies employed by Westinghouse are added. This retains the flexibility to resolve future emergent licensing issues using either analysis methodology when licensed to do so, as well as allowing the joint safety evaluations/analyses performed now and in the future. Additionally, commas are added within several dates for consistency, which are considered minor editorial changes.

Revisions to TS Bases B 2.1.1 deletes the value for enthalpy rise hot channel factor, revises the text to indicate that it is found in the COLR, and revises the equation to add a variable versus a fixed value. The changes reflects use of the Westinghouse RTDP analysis methodology.

Revision to TS Bases B 2.2.1 deletes the numerical value for DNBR that is stated in the section addressing Power Range, Neutron Flux, High Rates. The numerical DNBR value is replaced with a statement referencing the DNBR limits specified in the applicable NRC-approved analytical methods referenced in Specification 6.8.1.6.b. The change is based on approved Westinghouse DNB analysis methodologies using WRB-2, RTDP, and VIPRE.

Revisions to TS Bases B 3/4.2.2 and B 3/4.2.3 replaces the FIDS Bases discussion with F_Q discussion to reflect use of the approved Westinghouse RAOC TS.

References

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

[2] Davidson, S. L. (Ed.), et al., VANTAGE 5 Fuel Assembly Reference Core Report, WCAP-10444-P-A and Appendix A, September 1985; Addendum 1-A, March 1986; Addendum 2-A, April 1988.

[3] YAEC-1849P, Thermal-Hydraulic Analysis Methodology Using VIPRE-01 for PWR Applications, October 1992.

[4] WCAP-11397-P-A, (Proprietary), Revised Thermal Design Procedure, April 1989.

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

[6] 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.

[7] WCAP-14483-A, Rev. 0 (Non-Proprietary), "Generic Methodology for Expanding Core Operating Limits Report," January 1999.

[8] Letter from Byron L. Siegel (NRC) to D. N. Morey (Southern Nuclear Operating Company), Issuance of Amendments - Joseph M. Farley Nuclear Plant Units 1 and 2, (TAC Nos. M95700 and M95701), September 3, 1996.

Section II

Markup of the Proposed Changes

The attached markup reflects the currently issued revision of the Technical Specifications. Pending Technical Specification changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed markup.

The following Technical Specifications are included in the attached markup:

Technical Specification	Title	Page(s)
Figure 2.1-1	Reactor Core Safety Limit – Four Loops In Operation	2-2
Table 2.2-1	Reactor Trip System Instrumentation Trip Setpoints - Table Notations	2-7, 2-8 2-9, 2-10
B 2.1.1	Safety Limits Bases – Reactor Core	B 2-1
B 2.2.2	Limiting Safety System Settings – Reactor Trip System Instrumentation Setpoints -	B 2-4
3/4.2.1	Axial Flux Difference	3/4 2-1, 3/4 2-2
3/4.2.2	Heat Flux Hot Channel Factor – $F_Q(Z)$	3/4 2-4, 3/4 2-6
3/4.2.3	Nuclear Enthalpy Rise Hot Channel Factor	3/4 2-8
B 3/4.2.2 and B 3/4.2.3	Bases 3/4.2.2 and 3/4.2.3 Heat Flux Hot Channel Factor and Nuclear Enthalpy Rise Hot Channel Factor	B 3/4 2-3
6.8.1.6.b	Administrative Controls – Core Operating Limits Report	6-18A, 6-18B, 6-18C

REPLACE WITH ATTACHED NEW FIGURE 2.1-1

INSERT (1)

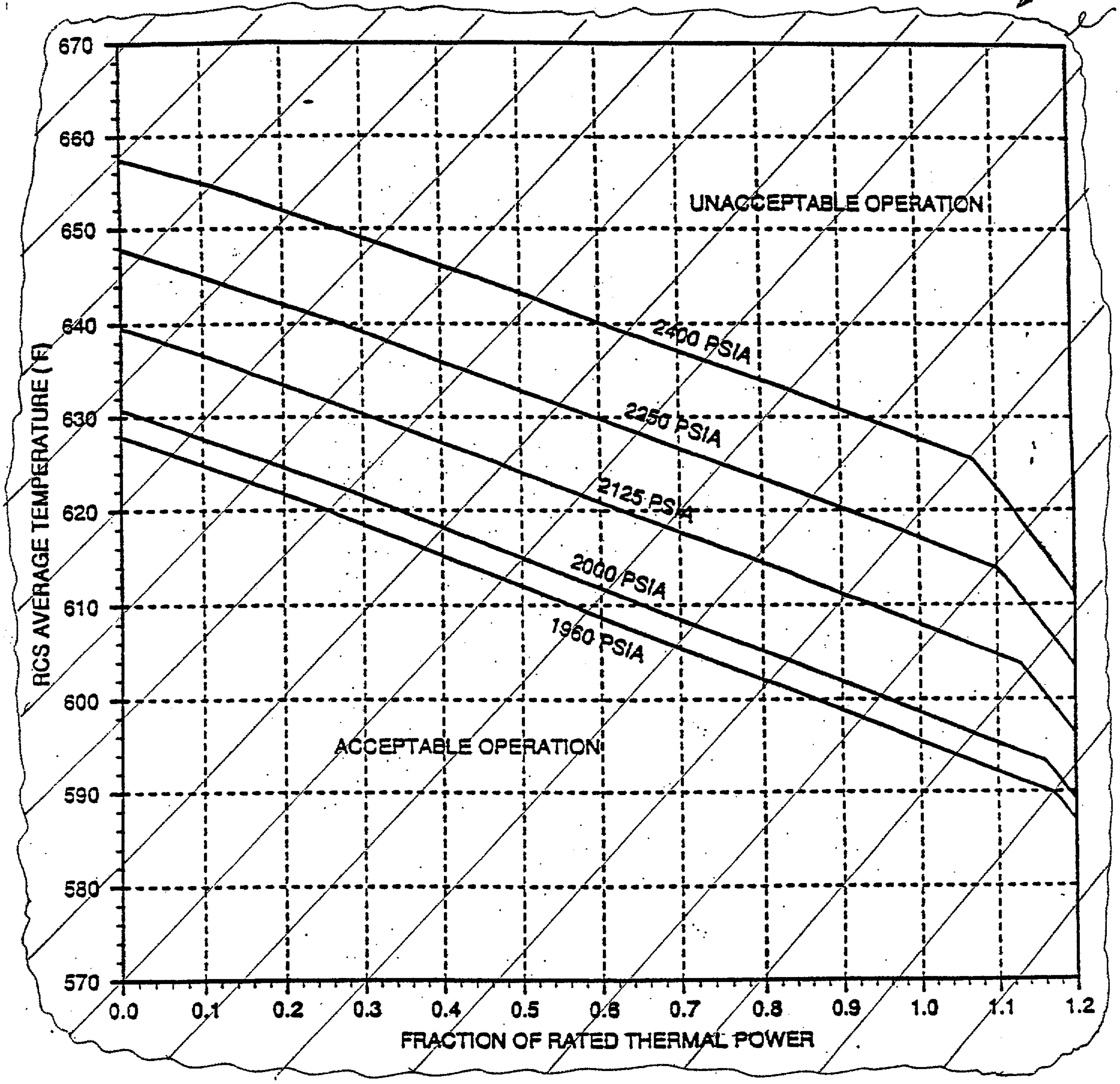

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

NEW FIGURE 2.1-1

INSERT 1

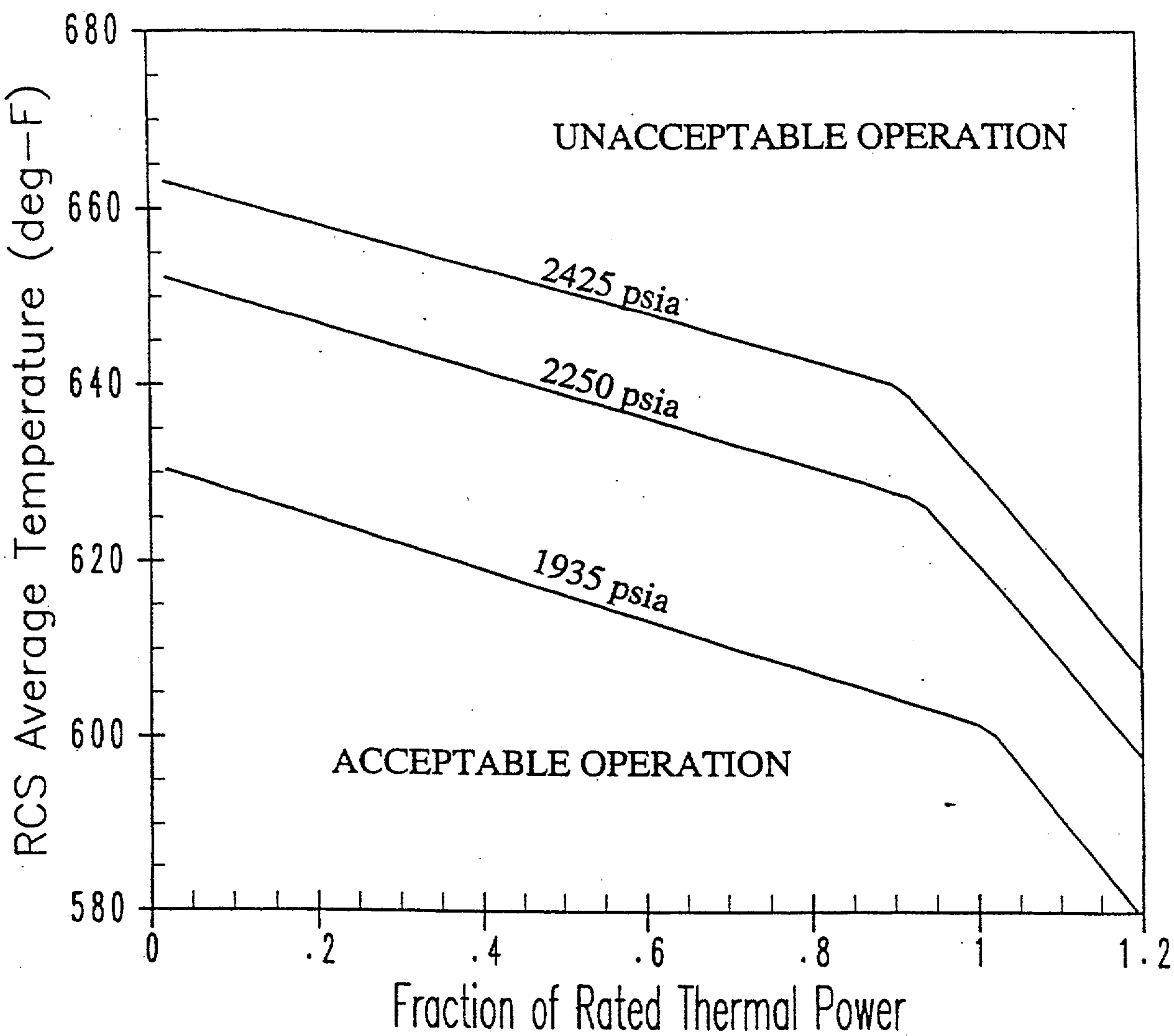

Figure 2.1-1

REACTOR CORE SAFETY LIMITS - FOUR LOOPS IN OPERATION

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \frac{(1)}{(1+\tau_3 S)} \le \Delta T_0 \{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} [T \frac{(1)}{(1+\tau_6 S)} - T^1] + K_3(P - P^1) - f_1(\Delta I)\}$$

Where:

ΔT = Measured RCS ΔT by RTD Instrumentation~~;~~, °F;

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

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, ~~$\tau_1 \geq 8$ s,~~ ~~$\tau_2 \leq 3$ s;~~ VALUES SPECIFIED IN THE COLR;

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

τ_3 = Time constants utilized in the lag compensator for ΔT, VALUE SPECIFIED IN THE COLR ~~$\tau_3 = 0$ s~~;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER~~;~~, °F;

K_1 = Value specified in the COLR;

K_2 = Value specified in the COLR;

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

τ_4, τ_5 = Time constants utilized in lead-lag compensator for T_{avg}, ~~$\tau_4 \geq 33$ s,~~ ~~$\tau_5 \leq 4$ s;~~ VALUES SPECIFIED IN THE COLR.

MEASURED RCS

T = Average temperature, °F;

$\frac{1}{1+\tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, VALUE SPECIFIED IN THE COLR ~~$\tau_6 = 0$ s~~;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

INDICATED RCS

T^1 ≤ 588.5°F (Nominal) T_{avg} at RATED THERMAL POWER; °F, (CALIBRATION TEMPERATURE FOR ΔT INSTRUMENTATION, VALUE SPECIFIED IN THE COLR);

K_3 = Value specified in COLR;

MEASURED

P = Pressurizer pressure, psig;

P^1 = 2235 psig (Nominal RCS operating pressure); , PSIG, VALUE SPECIFIED IN THE COLR;

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.

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

NOTE 3: OVERPOWER ΔT

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

MAKE this T'' SAME AS ON PAGE 2-10

Where:

ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 = Value specified in the COLR,

K_5 = Value specified in the COLR,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s, VALUE SPECIFIED IN THE COLR

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

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

NOTE 3: (Continued)

K_6 = Value specified in COLR,

T = As defined in Note 1,

T'' = Indicated T_{avg} at RATED THERMAL POWER °F (Calibration temperature for ΔT instrumentation, ~~≤ 588.5°F~~ OR VALUE SPECIFIED IN THE COLR),

S = As defined in Note 1, and

$f_2(\Delta I)$ = A function of the indicated difference between the top and bottom detectors of the power-range neutron ion chambers as specified in the COLR.

NOTE 4: 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.

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.

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

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

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.

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 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.

3/4.2.1 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.~~

(THIS SPECIFICATION NUMBER IS NOT USED)

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 (1)

$K(Z)$ = 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 - $F_Q(Z)$

SURVEILLANCE REQUIREMENTS

~~LIMITING CONDITION FOR OPERATION~~

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

INSERT 2

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:~~

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.~~

INSERT 2A

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:~~

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

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

4.2.2.4 ~~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.~~

(THIS SPECIFICATION NUMBER IS NOT USED)

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:

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

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

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

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) 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

2) $F_Q^M(Z)$ 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.

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 ② CONTINUED

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.

INSERT 2A

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 OR EQUAL TO the limits specified in the COL~~A~~R.

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 $F_Q^{(z)}$ 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.

~~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.

$F_Q(Z)$
SUBSCRIPT

The hot channel factor $F_Q^M(Z)$ 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 FQ(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.

INSERT 3

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)

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

INSERT 5

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.

INSERT 6

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.

INSERT 7

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.

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) →

7. 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 Specification~~s~~:
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.

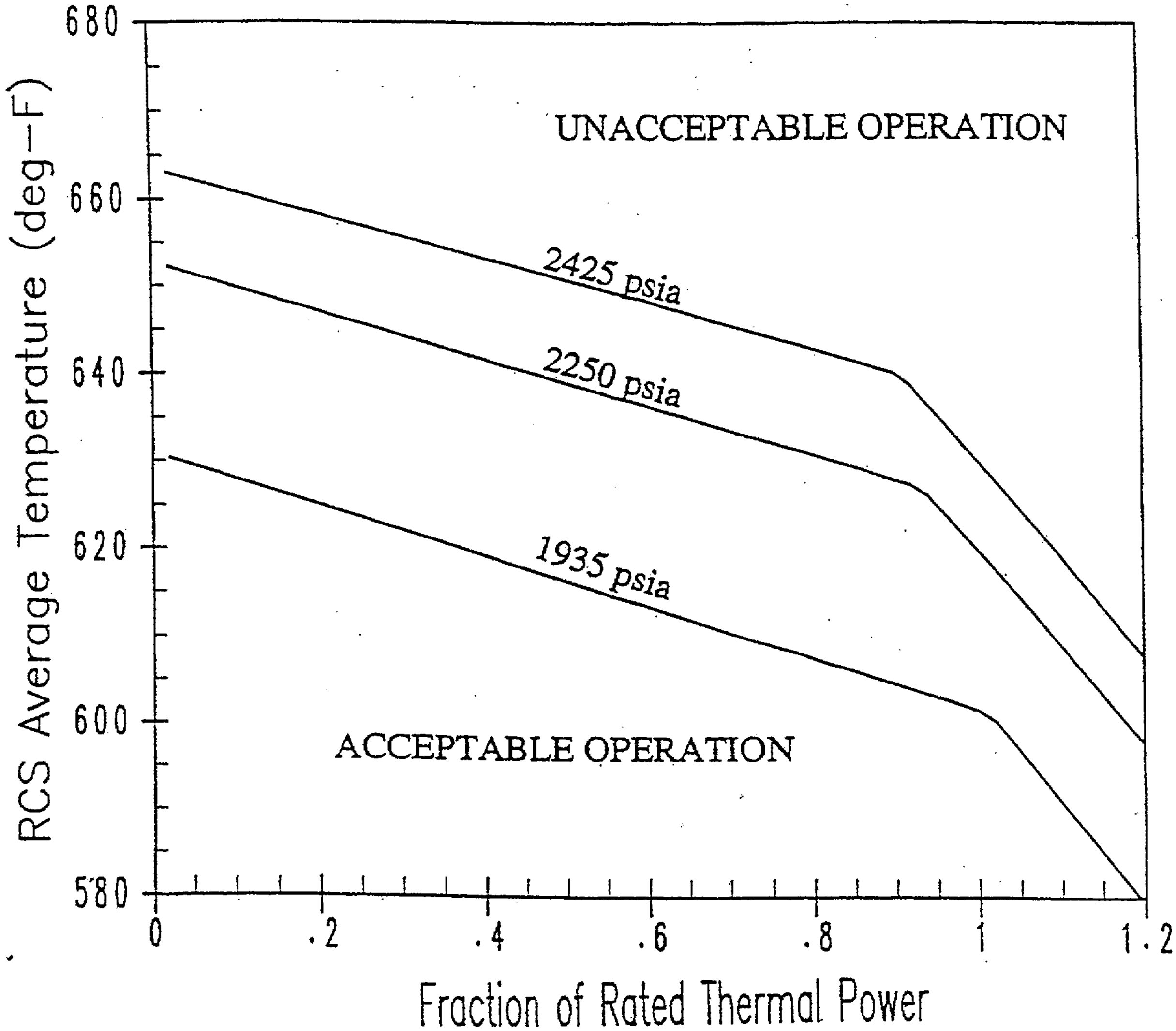

Figure 2.1-1

REACTOR CORE SAFETY LIMITS—FOUR LOOPS IN OPERATION

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \frac{(1)}{(1+\tau_3 S)} \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \frac{(1)}{(1+\tau_6 S)} - T^1\right] + K_3(P - P^1) - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured RCS ΔT by RTD Instrumentation, °F;
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, values specified in the COLR;
$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, value specified in the COLR;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER, °F;
K_1	=	Value specified in the COLR;
K_2	=	Value specified in the COLR;
$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in lead-lag compensator for T_{avg}, values specified in the COLR;
T	=	Measured RCS Average temperature, °F;
$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, value specified in the COLR;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

T^1 Indicated RCS T_{avg} at RATED THERMAL POWER, °F, (Calibration temperature for ΔT instrumentation, value specified in the COLR);

K_3 = Value specified in COLR;

P = Measured Pressurizer pressure, psig;

P^1 = Nominal RCS operating pressure, psig, value specified in the COLR;

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.

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

NOTE 3: OVERPOWER ΔT

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

Where: ΔT = As defined in Note 1,

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1+\tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 = Value specified in the COLR,

K_5 = Value specified in the COLR,

$\frac{\tau_7 S}{1+\tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in rate-lag compensator for T_{avg}, value specified in the COLR,

$\frac{1}{1+\tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

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

NOTE 3: (Continued)

K_6	=	Value specified in COLR,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER, °F, (Calibration temperature for ΔT instrumentation, value specified in the COLR),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	A function of the indicated difference between the top and bottom detectors of the power-range neutron ion chambers as specified in the COLR.

NOTE 4: 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.

3/4.2.1 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.

3/4.2.1 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 (THIS SPECIFICATION NUMBER IS NOT USED)

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) 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

2) $F_Q^M(Z)$ 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 (THIS SPECIFICATION NUMBER IS NOT USED)

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 $F_{\Delta H}^{RTP}$ 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

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:
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.

    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, Revision 1, 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).

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.

The proposed TS changes are:

- Figure 2.1-1 Revise Thermal Limit Lines.
- Table 2.2-1 Revise Table Notations to relocate additional cycle-specific parameters to the Core Operating Limits Report (COLR).
- 3.2.1 Delete Limiting Conditions for Operation associated with Fixed Incore Detection System (FIDS) Alarm.
- 3.2.1 Revise Action a.2. to delete reducing of Power Range Neutron Flux – High Trip Setpoints.
- 3.2.1 Delete Actions b. and c. associated with FIDS Alarm.
- 4.2.1.2 Delete Surveillance Requirement to determine maximum allowed power operation with FIDS Alarm inoperable.
- 3.2.2 Revise Action a.2. to delete identifying and correcting the cause of the out-of-limit condition.
- 4.2.2.2 Revise Surveillance Requirement to reflect F_Q methodology.
- 4.2.2.3 Revise Surveillance Requirement to reflect F_Q methodology.
- 4.2.2.4 Delete Surveillance Requirement to update FIDS Alarm setpoint every 31 days.
- 3.2.3 Editorial changes for consistency with the COLR.
- 3.2.3 Revise Action b. to delete identifying and correcting the cause of the out-of-limit condition.
- 6.8.1.6.b Updated to reflect approved methodology references.

In addition, page 3/4 2-6 is revised to correct the title, and the associated TS Bases are revised to reflect use of the Westinghouse Revised Thermal Design Procedure (RTDP) WRB-2 DNB correlation, and VIPRE modeling methodologies used in thermal-hydraulic and DNB analysis.

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.**

The proposed changes to TS 2.1.1, 3.2.1, 4.2.1.1, 4.2.2.2, 4.2.2.3, 4.2.2.4, 6.8.1.6.b, and changes to the aforementioned TS Bases, 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)). 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 Updated Final Safety Analysis Report (UFSAR).

The proposed changes to TS Table 2.2-1, TS 3.2.2, TS 3.2.3, and the title on page 3/4 2-6 are editorial changes to correct either typographical errors, simplification of statements, clarification of specific parameters associated with temperature / pressure measurements, making some notations consistent with improved Standard Technical Specifications – Westinghouse Plants, NUREG-1431, Rev. 1, and relocating additional cycle-specific values for temperature, pressure and time constants to the COLR, or correcting an erroneous title. These changes do not result in a change to the design basis of any plant structure, system or component or parameters currently specified in the COLR, therefore, operation of the facility within the prescribed limits of TS remains unchanged.

The proposed change to TS 3.2.1, ACTION a.2, to delete the need to reduce the power range neutron flux high trip setpoints subsequent to reducing rated thermal power (RTP) to less than 50% whenever axial flux difference (AFD) is outside of the applicable limits specified in the COLR, does not significantly increase the probability or consequences of an accident previously evaluated. Reducing the power level to less than or equal to 50 percent rated thermal power (RTP) maintains the plant in a benign condition since under RAOC methodology there are no AFD limits below 50 percent of RTP. In addition, a rapid rise in power to greater than 50 percent RTP with AFD outside limits does not immediately create an unacceptable situation. Since the transient analysis setpoint calculations for f (ΔI) (input to the overtemperature delta-temperature (OTΔT) trip function) are based on the same core power distributions that the fuel designers use for a reload cycle design, the OTΔT trip function provides an acceptable level of protection for such an excursion. Furthermore, the event would be successfully terminated by a trip at the previous setpoint level. The increased potential for a reactor trip caused by the manual manipulation of the setpoint needlessly exposes the plant to an unnecessary trip with the potential for an undesirable plant transient. Therefore, maintaining this provision as part of TS 3.2.1, Action a.2 is not warranted.

Therefore, for the reasons stated above, the probability or consequences of an accident previously evaluated are not significantly increased for all the proposed TS changes presented herein.

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 has shown that all design standards and applicable safety criteria continue to be met.

The proposed editorial changes and elimination for reducing the power range neutron flux high trip setpoint do not result in a change to the design basis of any plant structure, system or component. The level of protection afforded by these safety features is not affected by the proposed changes.

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 proposed changes are not expected to result in conditions that are more adverse 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.

The proposed editorial changes do not change the current limits specified in Technical Specifications. The current limits are based on safety analysis limits developed using NRC-approved methodologies specified in TS 6.8.1.6.

Removing the requirement for manually reducing the power range neutron flux high trip setpoint does not result in a significant reduction in a margin of safety. There are other levels of trip protection to terminate a rapid rise in power excursion, such as the overtemperature delta-temperature (OTΔT) trip function and previous power range neutron flux high trip setpoint. In addition, a rapid rise in power to greater than 50 percent RTP with AFD outside limits does not immediately create an unacceptable situation. The increased potential for a reactor trip caused by the manual manipulation of the setpoint needlessly exposes the plant to an unnecessary trip with the potential for an undesirable plant transient which may unnecessarily challenge safety systems.

Therefore, the proposed aforementioned TS 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, Revision 1, and issuance of a license amendment by October 15, 2000, having immediate effectiveness and implementation at commencement of Cycle 8 operation (currently scheduled mid-November, 2000)

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.