North Atlantic

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The Northeast Utilities System

April 15, 2002

Docket No. 50-443

NYN-02022

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Seabrook Station
License Amendment Request 01-10
"Relocation of Cycle-Specific Parameters to the Core Operating Limits Report"

North Atlantic Energy Service Corporation (North Atlantic) has enclosed herein License Amendment Request (LAR) 01-10. License Amendment Request 01-10 is submitted pursuant to the requirements of 10 CFR 50.90 and 10 CFR 50.4.

LAR 01-10 proposes changes to the Seabrook Station Technical Specifications (TS) to relocate boron concentration limits contained in certain TSs to the Core Operating Limits Report (COLR). Boron concentration qualifies as a cycle-specific parameter limit that is cycle dependent. The affected TS are: 3/4.1.1.1, "Boration Control Shutdown Margin – T_{avg} Greater Than or Equal To 200°F;" 3/4.1.1.2, "Boration Control Shutdown Margin – T_{avg} Less Than or Equal To 200°F;" 3/4.1.2.5, "Borated Water Sources – Shutdown;" 3/4.1.2.6, "Borated Water Sources – Operating;" 3/4.1.2.7, "Isolation of Unborated Water Sources;" 3/4.5.1.1, "Accumulators;" 3/4.5.4, "Refueling Water Storage Tank;" and 3/4.9.1, "Boron Concentration."

In addition, LAR 01-10 proposes changes to TS 2.1, "Safety Limits," to relocate Figure 2.1-1, "Reactor Core Safety Limits-Four Loops in Operation," to the COLR; and revise TSs 2.1.1 and 2.1.2 limiting conditions and actions so as to be consistent with the improved Standard Technical Specifications (ITS) – Westinghouse Plants, NUREG-1431, Revision 2. Also proposed is the relocation of DNB-related parameters, specified in TS 3/4.2.5, to the COLR.

As a result of the above changes, TS 6.8.1.6, "Core Operating Limits Report," and associated TS Bases must be revised to be reflective of the above changes. Furthermore, editorial and administrative changes to TS 6.8.1.6, consistent with ITS, are proposed as well.

The proposed changes are either based on (1) NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications", dated October 3, 1988; (2) the NRC staff's acceptance of WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report;" and/or (3) the improved Standard Technical Specifications, NUREG-1431, Revision 2.

A001

Relocation of cycle-specific parameters from TS to the COLR (a licensee-controlled document subject to the requirements of TS 6.8.1.6 and the provisions of 10 CFR 50.59) would afford North Atlantic flexibility to revise cycle-specific parameters, in accordance with NRC-approved methodologies, without the need for a license amendment. Specifically, TS 6.8.1.6c requires copies of the COLR to be submitted to the NRC for each reload cycle, including any mid-cycle revisions or supplements thereto. Thus resources, both North Atlantic and NRC, would be saved by minimizing and/or eliminating repetitive LAR submittals associated with revising cycle-specific parameters.

The Station Operation Review Committee and the Nuclear Safety Audit Review Committee have reviewed LAR 01-10.

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 01-10, and issuance of a license amendment by April 15, 2003 (see Section V enclosed).

North Atlantic has determined that LAR 01-10 meets the criterion 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.

Ted C. Feigenbaum
Executive Vice President
and Chief Nuclear Officer

cc:

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North Atlantic

SEABROOK STATION UNIT 1

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

License Amendment Request 01-10
"Relocation of Cycle-Specific Parameters to the Core Operating Limits Report"

North Atlantic Energy Service Corporation pursuant to 10 CFR 50.90 submits this License Amendment Request. 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, Ted C. Feigenbaum, Executive Vice President and Chief Nuclear Officer of North Atlantic Energy Service Corporation hereby affirm that the information and statements contained within this License Amendment Request are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Ted C. Feigenbaum
Executive Vice President
and Chief Nuclear Officer

Sworn and Subscribed
before me this
15th day of April, 2002

Notary Public

Section I

Introduction and Safety Assessment for the Proposed Change

I. INTRODUCTION AND SAFETY ASSESSMENT OF THE PROPOSED CHANGE

A. Introduction

License Amendment Request (LAR) 01-10 proposes changes to the Seabrook Station Technical Specifications (TS) to relocate boron concentration limits contained in certain TSs to the Core Operating Limits Report (COLR). Boron concentration qualifies as a cycle-specific parameter limit that is cycle dependent. The affected TS are: 3/4.1.1.1, "Boration Control Shutdown Margin – T_{avg} Greater Than or Equal To 200°F;" 3/4.1.1.2, "Boration Control Shutdown Margin – T_{avg} Less Than or Equal To 200°F;" 3/4.1.2.5, "Borated Water Sources – Shutdown;" 3/4.1.2.6, "Borated Water Sources – Operating;" 3/4.1.2.7, "Isolation of Unborated Water Sources;" 3/4.5.1.1, "Accumulators;" 3/4.5.4, "Refueling Water Storage Tank;" and 3/4.9.1, "Boron Concentration."

In addition, LAR 01-10 proposes changes to TS 2.1, "Safety Limits," to relocate Figure 2.1-1, "Reactor Core Safety Limits-Four Loops in Operation," to the COLR; and revise TSs 2.1.1 and 2.1.2 limiting conditions and actions so as to be consistent with the improved Standard Technical Specifications (ITS) – Westinghouse Plants, NUREG-1431, Revision 2. Also proposed is the relocation of DNB-related parameters, specified in TS 3/4.2.5, to the COLR.

As a result of the above changes, TS 6.8.1.6, Core Operating Limits Report, and associated TS Bases must be revised to be reflective of the above changes. Furthermore, editorial and administrative changes to TS 6.8.1.6, consistent with ITS, are proposed as well.

The proposed changes are either based on (1) NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications", dated October 3, 1988; (2) the NRC staff's acceptance of WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report;" and/or (3) the improved Standard Technical Specifications, NUREG-1431, Revision 2.

Relocation of cycle-specific parameters from TS to the COLR (a licensee-controlled document subject to the requirements of TS 6.8.1.6 and the provisions of 10 CFR 50.59) would afford North Atlantic flexibility to revise cycle-specific parameters, in accordance with NRC-approved methodologies, without the need for license amendment submittals. Specifically, TS 6.8.1.6c requires copies of the COLR to be submitted to the NRC for each reload cycle, including any mid-cycle revisions or supplements thereto. Thus resources, both North Atlantic and NRC, would be saved by minimizing and/or eliminating repetitive LAR submittals associated with revising cycle-specific parameters.

B. Safety Assessment

NRC Generic Letter 88-16 provided guidance for the preparation of a license amendment request to provide an alternative to identifying cycle-specific parameter limits within Technical Specifications. This alternative included three separate actions to modify the plant's Technical Specifications: (1) the addition of a definition of a named formal report that includes the values of cycle-specific parameter limits that have been established using an NRC-approved methodology that are consistent with all applicable limits of the safety analysis; (2) the addition of an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the Commission for information; and (3) the modification of individual Technical Specifications to note that the cycle-specific parameters shall be maintained within the limits provided in the defined report.

Amendment 9 to the Seabrook Station Operating License, dated February 18, 1992, authorized North Atlantic to implement the guidance of Generic Letter 88-16. Subsequently, other license amendments (e.g., Amendment 76, dated October 6, 2000) authorized relocation of certain additional cycle-specific parameters to the COLR. Relocation of boron concentration limits to the COLR conforms to the NRC methodology for expanding the COLR. Cycle-specific parameters, including borated water concentrations, are generated using NRC-approved methodologies. These methodologies are listed in Seabrook Station Technical Specification 6.8.1.6. Through issuance of Generic Letter 88-16, the NRC has determined that such cycle-specific variables may be removed from Technical Specifications and placed in a licensee-controlled Core Operating Limits Report; thus obviating prior NRC review and approval to facilitate changes. The COLR ensures that changes to the boron concentration limits will continue to be performed in accordance with NRC-approved methodologies used to derive the parameters. Thus, it is proposed that the COLR will contain all the TS-relocated boron concentration values, i.e., those values currently contained in the Limiting Condition for Operation (LCO), Action, and Surveillance Requirements[1]. Relocation of the values will afford North Atlantic operational flexibility to make changes to these values in accordance with the NRC-approved methodologies, as controlled by TS 6.8.1.6, without requiring a license amendment every time boron concentration values are changed.

In addition to the above proposed changes, it is proposed that limiting condition a. of TS 3/4.9.1 be deleted because it is redundant to condition b. insofar as the boron concentration value derived for condition b. is based on achieving a k_{eff} of 0.95 or less. This is currently stated in the associated Bases. Therefore, North Atlantic believes the Bases is the appropriate place for stating the basis for the boron concentration value while in Mode 6. Deletion of the k_{eff} value from the LCO would make this LCO consistent with ITS 3.9.1 which only specifies boron concentration.

The NRC Safety Evaluation to Topical Report WCAP-14483-A found relocation of DNB-related parameters (for TS 2.1.1, TS 3/4.2.5, and TS Figure 2.1-1) to the COLR, including the replacement of more specific requirements regarding the Safety Limits in TS 2.1.1 (i.e., the fuel DNB design basis and fuel centerline melt design basis), to be acceptable. Limitations on the combination of Thermal Power, pressurizer pressure and RCS temperature will be controlled by the COLR. The more specific Safety Limit requirements of fuel DNB design basis and fuel centerline melt design basis, as well as the minimum limit for Reactor Coolant System (RCS) total flow (based on that used in the safety analysis), will be controlled by the Technical Specifications.

With the incorporation of the more specific Safety Limits requirements in TS 2.1.1 North Atlantic is proposing to fashion TS 2.1.1 and 2.1.2 to be consistent with ITS 2.1 and 2.2, (NUREG-1431, Revision 2), including the associated Bases. In addition, actions associated with violation of Safety Limits will be consistent with ITS and will be numbered as new TS 2.1.3. Fashioning these TSs to be consistent with ITS would upgrade these TSs to current regulatory standards. The action requirements remain essentially the same with exception of referral to TS 6.6 for reporting/action requirements which ITS no longer requires. TS Task Force Traveler TSTF 5, Revision 1, justified the elimination of reporting/action requirements from ITS based on redundancy with 10 CFR 50.36, which specifies the reporting/action requirements for violation of Safety Limits.[2]

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[1] As a result of relocating the cycle-specific parameters for the affected LCOs, Action, and Surveillance Requirements, minor rearrangement of wording has been done for operator clarity.

[2] TS 6.6, "Safety Limit Violation," will be revised in a future administrative LAR submittal.

Surveillance Requirements for the aforementioned TSs will continue to ensure the verification of boron concentration and DNB-related parameters to be within the specified COLR limits.

The proposed changes to relocate cycle-specific Technical Specification parameter limits to the COLR will maintain adequate controls upon these parameters during normal plant operations and anticipated operational occurrences. The subject parameter limits will be administratively controlled in accordance with Technical Specification 6.8.1.6. Specifically, TS 6.8.1.6.c requires the COLR to be submitted to the NRC each reload cycle, including any mid-cycle revisions or supplements.

Proposed is an additional administrative change to TS 6.8.1.6a to delete the specific requirement for the COLR to be maintained available in the Control Room. North Atlantic regards this requirement as an unnecessary regulatory burden which is not consistent with ITS and does not meet the requirements of 10 CFR 50.36(c)(5) pertaining to administrative controls. As with other important documents for control room personnel use[3], North Atlantic will ensure the proper location of the COLR.

In conclusion, the cycle-specific parameter limits controlled by the subject Specifications do not need to be included within the scope of the Technical Specifications. The subject limits are adequately controlled by the COLR. Relocation of such cycle-specific limits from Technical Specifications to the COLR are consistent with the Commission's position established by Generic Letter 88-16, the Safety Evaluation of WCAP-14483-A, and the improved TS. Therefore, the proposed changes to the subject Technical Specifications, Index and Bases sections do not pose a significant hazard to the public health and safety.

[3] A specific technical specification does not exist requiring the TS manual (Appendix A to the Operating License) itself to be located in the Control Room.

Section II

Markup of Proposed Change

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

The following Technical Specifications and Bases are included in the attached markup:

Technical Specification	Title	Page(s)
Index Section 2.1	Safety Limits	ii
Specification 2.1	Safety Limits - Reactor Core and Reactor Coolant System Pressure	2-1
Figure 2.1-1	Reactor Core Safety Limits – Four Loops in Operation	2-2
Specification 3/4.1.1.1	Boration Control Shutdown Margin – Tavg Greater Than 200°F	3/4 1-1
Specification 3/4.1.1.2	Boration Control Shutdown Margin – Tavg Less Than 200°F	3/4 1-3
Specification 3/4.1.2.5	Borated Water Sources - Shutdown	3/4 1-11
Specification 3/4.1.2.6	Borated Water Sources - Operating	3/4 1-12
Specification 3/4.1.2.7	Isolation of Unborated Water Sources	3/4 1-14
Specification 3/4.2.5	DNB Parameters	3/4 2-10
Specification 3/4.5.1.1	Accumulators	3/4 5-1
Specification 3/4.5.4	Refueling Water Storage Tank	3/4 5-11
Specification 3/4.9.1	Boron Concentration	3/4.9-1
Specification 6.8.1.6.a & b	Core Operating Limits Report	6-18, 6-18A, B & E
Bases Specification 2.1.1	Reactor Core	B 2-1
Bases Specification 2.1.2	Reactor Coolant System Pressure	B 2-2
Bases Specification 2.2.1	Limiting safety System Settings (OPΔT)	B 2-5
Bases Specification 3/4.1.1	Boration Systems	B 3/4 1-1
Bases Specification 3/4.1.2	Boration Systems	B 3/4 1-3
Bases Specification 3/4.2.5	DNB Parameters	B 3/4 2-4
Bases Specification 3/4.9.1	Boron Concentration	B 3/4 9-1 & 9-2

2.1.3 SAFETY LIMIT VIOLATIONS 2-1

~~INDEX~~

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.0 BASES

3.0/4.0 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for four-loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.6.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.6.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.6.

INSERT A

AMENDMENT NO.

2B1 91 31

INSERT (A)

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS (SLs)

2.1.1 REACTOR CORE SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to 1.17 for the WRB-1/WRB-2/WRB-2M DNB correlations.

2.1.1.2 The peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained less than or equal to 2735 psig.

2.1.3 SAFETY LIMIT VIOLATIONS

2.1.3.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.1.3.2 If SL 2.1.2 is violated:

a. In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

b. In MODE 3, 4, or 5, restore compliance within 5 minutes.

(THIS FIGURE IS NOT USED)

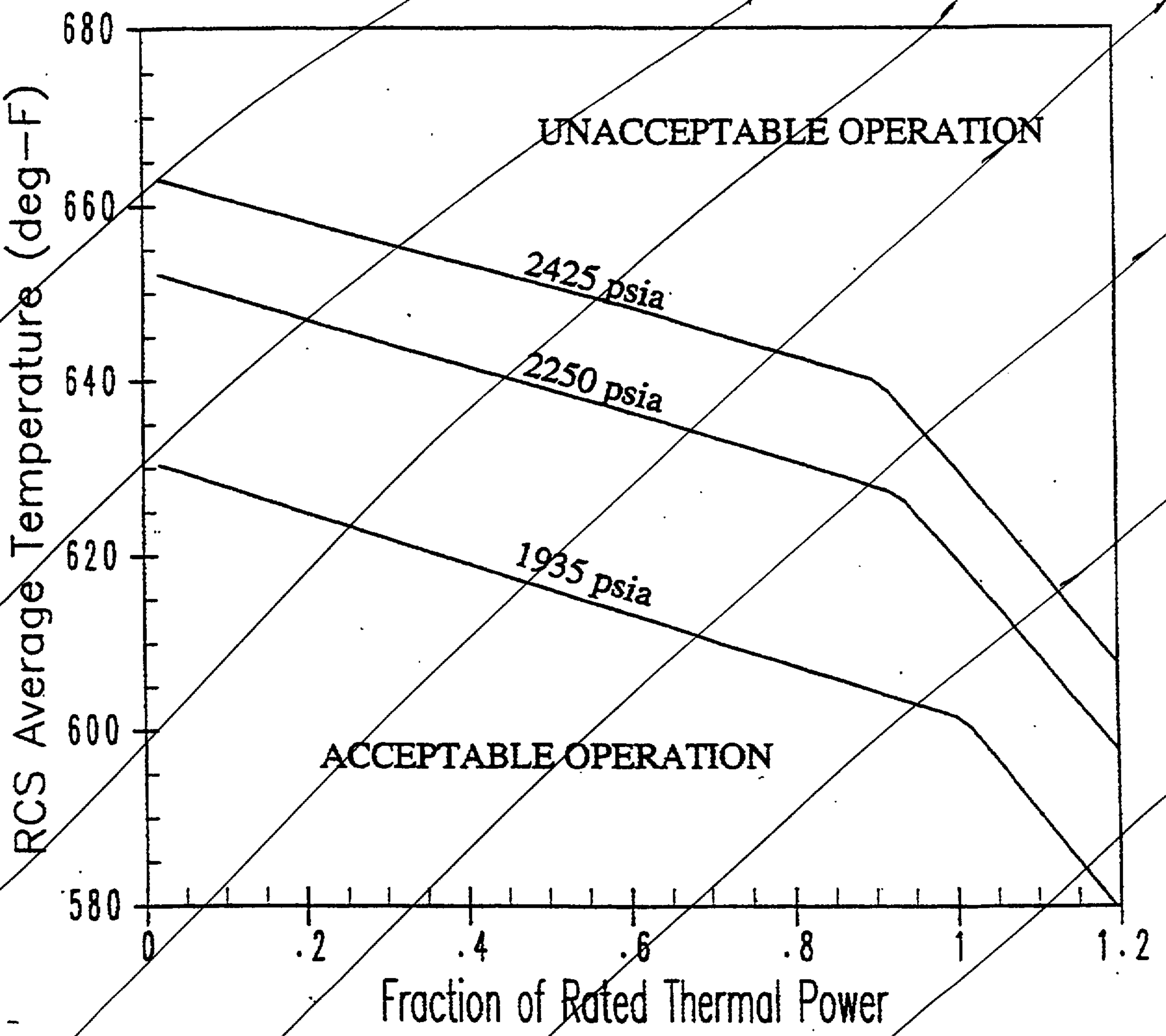

Figure 2.1-1

REACTOR CORE SAFETY LIMITS—FOUR LOOPS IN OPERATION

INSERT (B)

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.

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.

2.1.1 REACTOR CORE (Continued)

The resulting heat flux conditions are more limiting than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion, assuming the axial power imbalance is within the limits of the $f_1(\Delta I)$ and $f_2(\Delta I)$ functions of the Overtemperature and Overpower ΔT trips. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT and Overpower ΔT trips will reduce the setpoints to provide protection consistent with core safety limits for cycle specific power distribution.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The reactor vessel, pressurizer, and the RCS piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants, which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is, therefore, consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3110 psig) of design pressure to demonstrate integrity prior to initial operation.

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Continued

INSERT B

2.1.1 Reactor Core SLs

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. 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.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

INSERT B CONTINUED

2.1.1 Reactor Core SLs (continued)

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. Specification 3/4.2.5, "DNB Parameters," or the assumed initial conditions of the safety analyses (as indicated in the UFSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in Specification 3/4.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

2.1.2 Reactor Coolant System (RCS) Pressure SL

INSERT
B CONTINUED

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 3), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 3), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 4). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 5).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 6).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 3). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 2), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 2). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

INSERT (B) CONTINUED

2.1.2 Reactor Coolant System (RCS) Pressure SL (continued)

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs),

b. Steam line relief valve,

c. Steam Dump System,

d. Reactor Control System,

e. Pressurizer Level Control System, or

f. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 7) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

2.1.3 SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable:

If the reactor core SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable. The Allowed Outage Time (Completion Time) of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour. Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6). The Allowed Outage Time (Completion Time) of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

SAFETY LIMIT VIOLATIONS (continued)

INSERT B CONTINUED

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Chapter 7 AND 5 15.
3. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
4. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
5. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
6. 10 CFR 100.
7. USBS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), pressure is within the range between the Pressurizer High and Low Pressure trips and power is less than the Overpower ΔT trip setpoint. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in ~~Figure 2.1-1.~~ If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT   THE COLR.

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, and (3) axial power distribution to ensure that the allowable heat generation rate (Kw/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN for four-loop operation shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

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

ACTION:

EQUIVALENT

AT A BORON CONCENTRATION

With the SHUTDOWN MARGIN less than the limiting value, immediately initiate and continue boration ~~at greater than or equal~~ to 30 gpm ~~of a solution containing~~ greater than or equal ~~to 7000 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored.

to the limit specified in the COLR for the Boric Acid Storage System

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limiting value:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with k_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with k_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

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*See Special Test Exceptions Specification 3.10.1.

REACTIVITY CONTROL SYSTEMS

BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR). Additionally, the Reactor Coolant System boron concentration shall be greater than or equal ~~to 2000 ppm boron~~ The limit specified in The COLR when the reactor coolant loops are in a drained condition.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than the limit specified in the COLR or the Reactor Coolant System boron concentration less than ~~2000 ppm boron~~ The limit specified in The COLR, immediately initiate and continue boration ~~at greater than or equal~~ to 30 gpm ~~of a solution containing~~ AT A BORON CONCENTRATION greater than or equal to ~~7000 ppm boron or equivalent~~ THE LIMIT SPECIFIED IN THE COLR FOR THE BORIC ACID STORAGE SYSTEM OR EQUIVALENT until the required SHUTDOWN MARGIN and boron concentration are restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limit specified in the COLR and the Reactor Coolant System boron concentration shall be determined to be greater than or equal to ~~2000 ppm boron~~ THE LIMIT SPECIFIED IN THE COLR when the reactor coolant loops are in a drained condition:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

   1) Reactor Coolant System boron concentration,

   2) Control rod position,

   3) Reactor Coolant System average temperature,

   4) Fuel burnup based on gross thermal energy generation,

   5) Xenon concentration, and

   6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

BORATION SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 6,500 gallons,

AS SPECIFIED IN THE COLR

2) A minimum boron concentration ~~of 7000 ppm~~, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 24,500 gallons,

2) A minimum boron concentration ~~of 2700 ppm~~, and

3) A minimum solution temperature of 50°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water sources shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage System with:

AS SPECIFIED IN THE COLR

1) A minimum contained borated water volume of 22,000 gallons,
2) A minimum boron concentration ~~of 7000 ppm~~, and
3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

THE SPECIFIED LIMITS IN THE COLR

1) A minimum contained borated water volume of 477,000 gallons,
2) A boron concentration between ~~2700 and 2900 ppm~~,
3) A minimum solution temperature of 50°F, and
4) A maximum solution temperature of 98°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least the limit specified in the CORE OPERATING LIMITS REPORT (COLR) for the above MODES at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

REACTIVITY CONTROL SYSTEMS

BORON SYSTEMS

ISOLATION OF UNBORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 Provisions to isolate the Reactor Coolant System from unborated water sources shall be OPERABLE with:

a. The Boron Thermal Regeneration System (BTRS) isolated from the Reactor Coolant System, and

b. The Reactor Makeup Systems inoperable except for the capability of delivering up to the capacity of one Reactor Makeup Water pump to the Reactor Coolant System.

APPLICABILITY: MODES 4, 5, and 6

ACTION:

EQUIVALENT

With the requirements of the above specification not satisfied immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and, if within 1 hour the required SHUTDOWN MARGIN is not verified, initiate and continue boration ~~at greater than or equal~~ to 30 gpm ~~of a solution containing~~ greater than or equal to ~~7000 ppm boron or equivalent~~ THE LIMIT SPECIFIED IN THE COLR FOR THE BORIC ACID STORAGE SYSTEM until the required SHUTDOWN MARGIN is restored and the isolation provisions are restored to OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The provisions to isolate the Reactor Coolant System from unborated water sources shall be determined to be OPERABLE at least once per 31 days by:

a. Verifying that at least the BTRS outlet valve, CS-V-302, or the BTRS moderating heat exchanger outlet valve, CS-V-305, is closed and locked closed, and

b. Verifying that power is removed from at least one of the Reactor Makeup Water pumps, RMW-P-16A or RMW-P-16B.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

IS LESS THAN OR EQUAL TO THE LIMIT SPECIFIED IN THE COLR,

a. Reactor Coolant System T_{avg} ~~≤ 594.3°F~~

b. Pressurizer Pressure ~~≥ 2185 psig*~~

IS GREATER THAN OR EQUAL TO THE LIMIT SPECIFIED IN THE COLR*, AND

c. Reactor Coolant System Flow shall be:

1. ≥ 382,800 gpm**; and,

2. ≥ 392,800 gpm***

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits at least once per 12 hours.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by an approved method to be within its limit prior to operation above 95% of RATED THERMAL POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**Thermal Design Flow. An allowance for measurement uncertainty shall be made when comparing measured flow to Thermal Design Flow.

***Minimum measured flow used in the Revised Thermal Design Procedure.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

HOT STANDBY, STARTUP, AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6121 and 6596 gallons,

THE LIMITS SPECIFIED IN THE COLR

c. A boron concentration of between ~~2600 and 2900 ppm~~, and

d. A nitrogen cover-pressure of between 585 and 664 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With one pressure or water level channel inoperable per accumulator, return the inoperable channel to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With two pressure channels or two water level channels inoperable per accumulator, immediately declare the affected accumulator(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 24 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

*Pressurizer pressure above 1000 psig.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 477,000 gallons,

b. A boron concentration between ~~2700 and 2900 ppm of boron~~,

THE LIMITS SPECIFIED IN THE COLR

c. A minimum solution temperature of 50°F, and

d. A maximum solution temperature of 98°F.

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

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure ~~that the more restrictive of the following reactivity conditions is met; either:~~

a. ~~A k_{eff} of 0.95 or less, or~~

~~b.~~ ~~A~~ a boron concentration of greater than or equal to ~~2000 ppm~~ THE LIMIT SPECIFIED IN THE COLR.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or EQUIVALENT positive reactivity changes and initiate and continue boration ~~at greater than or equal~~ AT A BORON CONCENTRATION to 30 gpm ~~of a solution containing~~ greater than or equal to ~~7000 ppm boron or its equivalent~~ THE LIMIT SPECIFIED IN THE COLR FOR THE BORIC ACID STORAGE SYSTEM until ~~k_{eff} is reduced to less than or equal to 0.95 or~~ the boron concentration is restored to greater than or equal to ~~2000 ppm, whichever is the more restrictive.~~ THE LIMIT SPECIFIED IN THE COLR.

SURVEILLANCE REQUIREMENTS

4.9.1.1 ~~The more restrictive of the above two reactivity conditions shall be determined~~ VERIFY BORON CONCENTRATION IS WITHIN THE LIMITS SPECIFIED IN THE COLR prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience shall be submitted on monthly basis to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Document Control Desk, with a copy to the NRC Regional Administrator, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.8.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

1. Cycle dependent Overpower ΔT and Overtemperature ΔT trip setpoint parameters and function modifiers for operation with skewed axial power profiles for Table 2.2-1 of Specification 2.2.1,

~~2~~.3 SHUTDOWN MARGIN AND MINIMUM BORON CONCENTRATION limits for MODES 1, 2, 3, and 4 for Specification 3.1.1.1,

~~3~~.4 SHUTDOWN MARGIN AND MINIMUM BORON CONCENTRATION limits for MODE 5 for Specification 3.1.1.2,

~~4~~.5 Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3.1.1.3,

2. CYCLE DEPENDENT MAXIMUM ALLOWABLE COMBINATION OF THERMAL POWER, PRESSURIZER PRESSURE AND THE HIGHEST OPERATING LOOP ~~COOLANT~~ AVERAGE TEMPERATURE (TAVG) FOR SPECIFICATIONS 2.1.1 AND 2.1.2,

6. THE MINIMUM ~~[illegible]~~ BORON CONCENTRATION FOR MODES 5 AND 6 FOR SPECIFICATION 3.1.2.5,

7. THE ~~MINIMUM~~ BORON CONCENTRATION LIMITS FOR MODES 1, 2, 3 AND 4 ~~4, 5 AND 6~~ FOR SPECIFICATION 3.1.2.6,

8. THE MINIMUM BORON CONCENTRATION FOR MODES 4, 5 AND 6 FOR SPECIFICATION 3.1.2.7,

6.8.1.6.a. (Continued)

9 ~~5~~. Shutdown Rod Insertion limit for Specification 3.1.3.5,

10 ~~6~~. Control Rod Bank Insertion limits for Specification 3.1.3.6,

11 ~~7~~. AXIAL FLUX DIFFERENCE limits for Specification 3.2.1,

12 ~~8~~. Heat Flux Hot Channel Factor, F_Q^{RTP} and K(Z) for Specification 3.2.2,

13 ~~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, Rev. 2 with Addenda (Nonproprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March, 1987.

   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.

14. CYCLE DEPENDENT DNB-RELATED PARAMETERS FOR REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE (TAVG), AND PRESSURIZER PRESSURE FOR SPECIFICATION 3.2.5

15. THE BORON CONCENTRATION LIMITS FOR MODES 1, 2, AND 3 FOR SPECIFICATION 3.5.1.1

16. THE BORON CONCENTRATION LIMITS FOR MODES 1, 2, 3, AND 4 FOR SPECIFICATION 3.5.4

17. THE BORON CONCENTRATION LIMITS FOR MODE 6 FOR SPECIFICATION 3.9.1.

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.

2.1 SAFETY LIMITS

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
3.2.5 - DNB PARAMETERS

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.

Add after letter

WCAP-15025P, "MODIFIED WRB-2 CORRELATION, WRB-2M, FOR PREDICTING CRITICAL HEAT FLUX IN 17X17 ROD BUNDLES WITH MODIFIED LPD MIXING VANE GRIDS," FEBRUARY 1998.

6.8.1.6.b. (Continued)

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

2.1 SAFETY LIMITS

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.

3.2.5 - DNB PARAMETERS
3.5.1.1 - ACCUMULATORS FOR MODES 1, 2 AND 3
3.5.4 - REFUELING WATER STORAGE TANK FOR MODES 1, 2, 3, AND 4
3.9.1 - BORON CONCENTRATION

3.1.2.5 - BORATED WATER SOURCES - SHUTDOWN
3.1.2.6 - BORATED WATER SOURCES - OPERATING
3.1.2.7 - ISOLATION OF UNBORATED WATER SOURCES - SHUTDOWN

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no-load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR) is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200° F, the reactivity transients resulting from a postulated steam line break cooldown are minimal. A SHUTDOWN MARGIN as specified in the COLR and a boron concentration of greater than ~~2000 ppm~~ THE LIMIT SPECIFIED IN THE COLR are required to permit sufficient time for the operator to terminate an inadvertent boron dilution event with T_{avg} less than 200° F. ①

INSERT ⓒ

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting end of cycle life (EOL) MTC value as specified in the COLR. The 300 ppm surveillance limit MTC value as specified in the COLR represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value as specified in the COLR.

LCO 3.1.2.6 SPECIFIES THE MINIMUM REQUIREMENTS FOR BORATED WATER SOURCES AVAILABILITY DURING MODES 1 THROUGH 4.

3/4.1.2 BORATION SYSTEMS (Continued)

boron capability requirement occurs at EOL from full power equilibrium xenon conditions. ~~and requires 22,000 gallons of 7000 ppm borated water from the boric acid storage tanks or a minimum contained volume of 477,000 gallons of 2700 - 2900 ppm borated water from the refueling water storage tank (RWST).~~

INSERT D

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable in MODES 4, 5, and 6 except when the reactor vessel head closure bolts are fully detensioned or the vessel head is removed, provides assurance that a mass addition pressure transient can be relieved by operation of a single PORV or an RHR suction relief valve.

As a result of this, only one boron injection system is available. This is acceptable on the basis of the stable reactivity condition of the reactor, the emergency power supply requirement for the OPERABLE charging pump and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT after xenon decay and cooldown from 200° F to 140° F. ~~This condition requires a minimum contained volume of 6500 gallons of 7000 ppm borated water from the boric acid storage tanks or a minimum contained volume of 24,500 gallons of 2700 ppm borated water from the RWST.~~

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limitations on OPERABILITY of isolation provisions for the Boron Thermal Regeneration System and the Reactor Water Makeup System in Modes 4, 5, and 6 ensure that the boron dilution flow rates cannot exceed the value assumed in the transient analysis.

LCO 3.1.2.5 SPECIFIES THE MINIMUM REQUIREMENTS FOR BORATED WATER SOURCES AVAILABILITY DURING MODES 5 AND 6.

INSERT
C

The "equivalent to" statement in the Action is a provision providing an alternate method of emergency boration via the RWST at an increased flow rate to account for the lower boron concentration within the RWST.

INSERT
D

The "equivalent to" statement in the Action for LCO 3.1.2.7 is a provision providing an alternate method of emergency boration via the RWST at an increased flow rate to account for the lower boron concentration within the RWST.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the updated FSAR assumptions and have been analytically demonstrated adequate to assure compliance with acceptance criteria for each analyzed transient. Operating procedures include allowances for measurement and indication uncertainty so that the limits of 594.3°F for T_{avg} and 2185 psig for pressurizer pressure are not exceeded. SPECIFIED IN THE COLR

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the specified limit.

RCS flow must be greater than or equal to, 1) the Thermal Design Flow (TDF) with an allowance for measurement uncertainty and, 2) the minimum measured flow used in place of the TDF in the analysis of the DNB related events when the Revised Thermal Design Procedure (RTDP) methodology is utilized. Measurement of RCS total flow rate is performed by performance of either a precision calorimetric heat balance or normalized cold leg elbow tap ΔP measurements. RCS flow measurements using either the precision heat balance or the elbow tap ΔP measurement methods are to be performed at steady state conditions prior to operation above 95% rated thermal power (RTP) at the beginning of a new fuel cycle. The elbow tap RCS flow measurement methodology is described in WCAP-15404, "Justification of Elbow Taps for RCS Flow Verification at Seabrook Station", dated April 2000.

3/4.9.1 BORON CONCENTRATION

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal/cavity during refueling ensures that the reactor remains subcritical during MODE 6. During refueling, the spent fuel pool water volumes and the reactor cavity water volumes will be connected when the fuel transfer gate valve is open. This configuration allows the bodies of water to be physically capable of being in contact, however, no effective mixing of the volumes occurs due to the constriction of the fuel transfer tube. The soluble boron concentration in each of these volumes is maintained greater than or equal to ~~2000 ppm boron~~ or equivalent to a K_{eff} less than or equal to 0.95 when the fuel transfer gate is open. However, the spent fuel pool water boron concentration is under administrative controls and not a technical specification. They are independently maintained at the appropriate boron concentration even though no intermixing of significance exists. The mixing caused by the RHR pumps (reactor cavity) or the SFP system pumps assures uniformity of boron in the separate volumes.

The limit as specified in the COLR,

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided. One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added solution of boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.8.1, "Residual Heat Removal (RHR) and Coolant Circulation High Water Level," and LCO 3.9.8.2, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling canal/cavity at or above the limit specified in LCO 3.9.1.

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least 5% $\Delta k/k$ margin of safety is established during refueling.

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO.

During refueling operations water may be transferred to the refueling canal/cavity or the RCS from different sources. Transfers or additions of water whose boron concentration exceeds the required refueling boron concentration are acceptable. Transfers or additions of water where the boron concentration is less than the required refueling boron concentration may be made, provided that these additions are administratively controlled to ensure that the refueling boron concentration requirements continue to be met. That is, the final concentration of boron in the total volume, after the addition of water less than the required refueling boron concentration, exceeds the required refueling boron concentration, or $k_{eff} \leq 0.95$. Also, these administrative controls ensure such transfers or additions of water will not substantially reduce the uniformity of boron concentration in the RCS or refueling canal.

Likewise, transferring water to the RCS or the refueling canal/cavity that is lower in temperature (down to the operability requirements of the RWST in MODE 6; 50 DEG F) than the water contained in those volumes is also acceptable. These minimum requirements for boron concentration and water temperature are also applicable to other MODE 6 Technical Specification ACTIONS that limit operations involving positive reactivity additions to ensure that the reactor remains subcritical and an adequate shutdown margin is maintained.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions. INSERT 1

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

Section III

Retype of 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 Technical Specifications prior to issuance.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECTION PAGE

BASES

3.0/4.0 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

2.1 SAFETY LIMITS (SLs)

2.1.1 REACTOR CORE SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to 1.17 for the WRB-1/WRB-2/WRB-2M DNB correlations.

2.1.1.2 The peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained less than or equal to 2735 psig.

2.1.3 SAFETY LIMIT VIOLATIONS

2.1.3.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.1.3.2 If SL 2.1.2 is violated:

a. In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

b. In MODE 3, 4, or 5, restore compliance within 5 minutes.

Figure 2.1-1 (THIS FIGURE IS NOT USED)

2.1.1 Reactor Core SLs

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. 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.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

2.1.1 Reactor Core SLs (continued)

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. Specification 3/4.2.5, "DNB Parameters," or the assumed initial conditions of the safety analyses (as indicated in the UFSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in Specification 3/4.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

2.1.2 Reactor Coolant System (RCS) Pressure SL

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 3), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 3), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 4). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be
pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 5).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 6).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 3). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 2), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 2). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

2.1.2 Reactor Coolant System (RCS) Pressure SL (continued)

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs),

b. Steam line relief valve,

c. Steam Dump System,

d. Reactor Control System,

e. Pressurizer Level Control System, or

f. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 7) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

2.1.3 SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable:

If the reactor core SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable. The Allowed Outage Time (Completion Time) of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour. Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6). The Allowed Outage Time (Completion Time) of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

SAFETY LIMIT VIOLATIONS (continued)

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Chapters 7 and 15.
3. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
4. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
5. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
6. 10 CFR 100.
7. USBS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), pressure is within the range between the Pressurizer High and Low Pressure trips and power is less than the Overpower ΔT trip setpoint. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in the COLR. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, and (3) axial power distribution to ensure that the allowable heat generation rate (Kw/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{AVG} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN for four-loop operation shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

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

ACTION:

With the SHUTDOWN MARGIN less than the limiting value, immediately initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limiting value:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with k_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with k_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

REACTIVITY CONTROL SYSTEMS

BORATION CONTROL

SHUTDOWN MARGIN -T_{AVG} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR). Additionally, the Reactor Coolant System boron concentration shall be greater than or equal to the limit specified in the COLR when the reactor coolant loops are in a drained condition.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than the limit specified in the COLR or the Reactor Coolant System boron concentration less than the limit specified in the COLR, immediately initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN and boron concentration are restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limit specified in the COLR and the Reactor Coolant System boron concentration shall be determined to be greater than or equal to the limit specified in the COLR when the reactor coolant loops are in a drained condition:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

   1) Reactor Coolant System boron concentration,

   2) Control rod position,

   3) Reactor Coolant System average temperature,

   4) Fuel burnup based on gross thermal energy generation,

   5) Xenon concentration, and

   6) Samarium concentration.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 6,500 gallons,

2) A minimum boron concentration as specified in the COLR, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 24,500 gallons,

2) A minimum boron concentration as specified in the COLR, and

3) A minimum solution temperature of 50°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water sources shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 22,000 gallons,

2) A minimum boron concentration as specified in the COLR, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 477,000 gallons,

2) A boron concentration between the specified limits in the COLR,

3) A minimum solution temperature of 50°F, and

4) A maximum solution temperature of 98°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least the limit specified in the CORE OPERATING LIMITS REPORT (COLR) for the above MODES at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ISOLATION OF UNBORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 Provisions to isolate the Reactor Coolant System from unborated water sources shall be OPERABLE with:

a. The Boron Thermal Regeneration System (BTRS) isolated from the Reactor Coolant System, and

b. The Reactor Makeup Systems inoperable except for the capability of delivering up to the capacity of one Reactor Makeup Water pump to the Reactor Coolant System.

APPLICABILITY: MODES 4, 5, and 6

ACTION:

With the requirements of the above specification not satisfied immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and, if within 1 hour the required SHUTDOWN MARGIN is not verified, initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN is restored and the isolation provisions are restored to OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The provisions to isolate the Reactor Coolant System from unborated water sources shall be determined to be OPERABLE at least once per 31 days by:

a. Verifying that at least the BTRS outlet valve, CS-V-302, or the BTRS moderating heat exchanger outlet valve, CS-V-305, is closed and locked closed, and

b. Verifying that power is removed from at least one of the Reactor Makeup Water pumps, RMW-P-16A or RMW-P-16B.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System T_{avg} is less than or equal to the limit specified in the COLR,

b. Pressurizer Pressure is greater than or equal to the limit specified in the COLR*, and

c. Reactor Coolant System Flow shall be:

1. $\geq$ 382,800 gpm**; and,

2. $\geq$ 392,800 gpm***

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits at least once per 12 hours.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by an approved method to be within its limit prior to operation above 95% of RATED THERMAL POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

** Thermal Design Flow. An allowance for measurement uncertainty shall be made when comparing measured flow to Thermal Design Flow.

*** Minimum measured flow used in the Revised Thermal Design Procedure.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

HOT STANDBY, STARTUP, AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6121 and 6596 gallons,

c. A boron concentration of between the limits specified in the COLR, and |

d. A nitrogen cover-pressure of between 585 and 664 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With one pressure or water level channel inoperable per accumulator, return the inoperable channel to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With two pressure channels or two water level channels inoperable per accumulator, immediately declare the affected accumulator(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 24 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

*Pressurizer pressure above 1000 psig.

BORON INJECTION SYSTEM

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 477,000 gallons,

b. A boron concentration between the limits specified in the COLR,

c. A minimum solution temperature of 50°F, and

d. A maximum solution temperature of 98°F.

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

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure a boron concentration of greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the boron concentration is restored to greater than or equal to the limit specified in the COLR.

SURVEILLANCE REQUIREMENTS

4.9.1.1 Verify boron concentration is within the limits specified in the COLR prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience shall be submitted on monthly basis to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Document Control Desk, with a copy to the NRC Regional Administrator, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.8.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

1. Cycle dependent Overpower ΔT and Overtemperature ΔT trip setpoint parameters and function modifiers for operation with skewed axial power profiles for Table 2.2-1 of Specification 2.2.1,

2. Cycle dependent maximum allowable combination of thermal power, pressurizer pressure and the highest operating loop average temperature (T_{avg}) for Specifications 2.1.1 and 2.1.2,

3. SHUTDOWN MARGIN and minimum boron concentration limits for MODES 1, 2, 3, and 4 for Specification 3.1.1.1,

4. SHUTDOWN MARGIN and minimum boron concentration limits for MODE 5 for Specification 3.1.1.2,

5. Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3.1.1.3,

6. The minimum boron concentration for MODES 5 and 6 for Specification 3.1.2.5,

7. The boron concentration limits for Modes 1, 2, 3, and 4 for Specification 3.1.2.6,

8. The minimum boron concentration for Modes 4, 5, and 6 for Specification 3.1.2.7,

9. Shutdown Rod Insertion limit for Specification 3.1.3.5,

10. Control Rod Bank Insertion limits for Specification 3.1.3.6,

11. AXIAL FLUX DIFFERENCE limits for Specification 3.2.1,

6.8.1.6.a. (Continued)

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

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

14. Cycle dependent DNB-related parameters for reactor coolant system average temperature (T_{avg}), and pressurizer pressure for Specification 3.2.5,

15. The boron concentration limits for MODES 1, 2 and 3 for Specification 3.5.1.1,

16. The boron concentration limits for MODES 1, 2, 3 and 4 for Specification 3.5.4,

17. The boron concentration limits for MODE 6 for Specification 3.9.1.

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, Rev. 2 with Addenda (Nonproprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March, 1987.

   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.

WCAP-15025-P, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17X17 Rod Bundles with Modified LPD Mixing Vane Grids," February 1998.

Methodology for Specification:
2.1 - Safety Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor
3.2.5 - DNB Parameters

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)

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

Methodology for Specifications:

2.1	-	Safety Limits
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
3.1.2.5	-	Borated Water Sources – Shutdown
3.1.2.6	-	Borated Water Sources – Operating
3.1.2.7	-	Isolation of Unborated Water Sources - Shutdown
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
3.2.5	-	DNB Parameters
3.5.1.1	-	Accumulators for MODES 1, 2 and 3
3.5.4	-	Refueling Water Storage Tank for MODES 1, 2, 3, and 4
3.9.1	-	Boron Concentration

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.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no-load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR) is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal. A SHUTDOWN MARGIN as specified in the COLR and a boron concentration of greater than the limit specified in the COLR are required to permit sufficient time for the operator to terminate an inadvertent boron dilution event with T_{avg} less than 200°F.

The “equivalent to” statement in the Action is a provision providing an alternate method of emergency boration via the RWST at an increased flow rate to account for the lower boron concentration within the RWST.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting end of cycle life (EOL) MTC value as specified in the COLR. The 300 ppm surveillance limit MTC value as specified in the COLR represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value as specified in the COLR.

3/4.1.2 BORATION SYSTEMS (Continued)

boron capability requirement occurs at EOL from full power equilibrium xenon conditions. LCO 3.1.2.6 specifies the minimum requirements for borated water sources availability during MODES 1, 2, 3 and 4. The "equivalent to" statement in the Action for LCO 3.1.2.7 is a provision providing an alternate method of emergency boration via the RWST at an increased flow rate to account for the lower boron concentration within the RWST.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable in MODES 4, 5, and 6 except when the reactor vessel head closure bolts are fully detensioned or the vessel head is removed, provides assurance that a mass addition pressure transient can be relieved by operation of a single PORV or an RHR suction relief valve.

As a result of this, only one boron injection system is available. This is acceptable on the basis of the stable reactivity condition of the reactor, the emergency power supply requirement for the OPERABLE charging pump and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT after xenon decay and cooldown from 200°F to 140°F. LCO 3.1.2.5 specifies the minimum requirements for borated water sources availability during MODES 5 and 6.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limitations on OPERABILITY of isolation provisions for the Boron Thermal Regeneration System and the Reactor Water Makeup System in Modes 4, 5, and 6 ensure that the boron dilution flow rates cannot exceed the value assumed in the transient analysis.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the updated FSAR assumptions and have been analytically demonstrated adequate to assure compliance with acceptance criteria for each analyzed transient. Operating procedures include allowances for measurement and indication uncertainty so that the limits specified in the COLR for T_{avg} and for pressurizer pressure are not exceeded.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the specified limit.

RCS flow must be greater than or equal to, 1) the Thermal Design Flow (TDF) with an allowance for measurement uncertainty and, 2) the minimum measured flow used in place of the TDF in the analysis of the DNB related events when the Revised Thermal Design Procedure (RTDP) methodology is utilized. Measurement of RCS total flow rate is performed by performance of either a precision calorimetric heat balance or normalized cold leg elbow tap ΔP measurements. RCS flow measurements using either the precision heat balance or the elbow tap ΔP measurement methods are to be performed at steady state conditions prior to operation above 95% rated thermal power (RTP) at the beginning of a new fuel cycle. The elbow tap RCS flow measurement methodology is described in WCAP-15404, "Justification of Elbow Taps for RCS Flow Verification at Seabrook Station", dated April 2000.

3/4.9.1 BORON CONCENTRATION

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal/cavity during refueling ensures that the reactor remains subcritical during MODE 6. During refueling, the spent fuel pool water volumes and the reactor cavity water volumes will be connected when the fuel transfer gate valve is open. This configuration allows the bodies of water to be physically capable of being in contact, however, no effective mixing of the volumes occurs due to the constriction of the fuel transfer tube. The soluble boron concentration in each of these volumes is maintained greater than or equal to the limit specified in the COLR, or equivalent to a K_{eff} less than or equal to 0.95 when the fuel transfer gate is open. However, the spent fuel pool water boron concentration is under administrative controls and not a technical specification. They are independently maintained at the appropriate boron concentration even though no intermixing of significance exists. The mixing caused by the RHR pumps (reactor cavity) or the SFP system pumps assures uniformity of boron in the separate volumes.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided. One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added solution of boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.8.1, "Residual Heat Removal (RHR) and Coolant Circulation High Water Level," and LCO 3.9.8.2, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling canal/cavity at or above the limit specified in LCO 3.9.1.

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least 5% $\Delta k/k$ margin of safety is established during refueling.

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO.

During refueling operations water may be transferred to the refueling canal/cavity or the RCS from different sources. Transfers or additions of water whose boron concentration exceeds the required refueling boron concentration are acceptable. Transfers or additions of water where the boron concentration is less than the required refueling boron concentration may be made, provided that these additions are administratively controlled to ensure that the refueling boron concentration requirements continue to be met. That is, the final concentration of boron in the total volume, after the addition of water less than the required refueling boron concentration, exceeds the required refueling boron concentration, or $k_{eff} \leq 0.95$. Also, these administrative controls ensure such transfers or additions of water will not substantially reduce the uniformity of boron concentration in the RCS or refueling canal.

Likewise, transferring water to the RCS or the refueling canal/cavity that is lower in temperature (down to the operability requirements of the RWST in MODE 6; 50 DEG F) than the water contained in those volumes is also acceptable. These minimum requirements for boron concentration and water temperature are also applicable to other MODE 6 Technical Specification ACTIONS that limit operations involving positive reactivity additions to ensure that the reactor remains subcritical and an adequate shutdown margin is maintained.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions. The "equivalent to" statement in the Action is a provision providing an alternate method of emergency boration via the RWST at an increased flow rate to account for the lower boron concentration within the RWST.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

Section IV

Determination of Significant Hazards for the Proposed Change

IV DETERMINATION OF SIGNIFICANT HAZARDS FOR THE PROPOSED CHANGE

License Amendment Request (LAR) 01-10 proposes changes to the Seabrook Station Technical Specifications (TS) to relocate boron concentration limits contained in certain TSs to the Core Operating Limits Report (COLR). Boron concentration qualifies as a cycle-specific parameter limit that is cycle dependent. The affected TS are: 3/4.1.1.1, "Boration Control Shutdown Margin – T_{avg} Greater Than or Equal To 200°F;" 3/4.1.1.2, "Boration Control Shutdown Margin – T_{avg} Less Than or Equal To 200°F;" 3/4.1.2.5, "Borated Water Sources – Shutdown;" 3/4.1.2.6, "Borated Water Sources – Operating;" 3/4.1.2.7, "Isolation of Unborated Water Sources;" 3/4.5.1.1, "Accumulators;" 3/4.5.4, "Refueling Water Storage Tank;" and 3/4.9.1, "Boron Concentration."

In addition, LAR 01-10 proposes changes to TS 2.1, "Safety Limits," to relocate Figure 2.1-1, "Reactor Core Safety Limits-Four Loops in Operation," to the COLR; and revise TSs 2.1.1 and 2.1.2 limiting conditions and actions so as to be consistent with the improved Standard Technical Specifications (ITS) – Westinghouse Plants, NUREG-1431, Revision 2. Also proposed is the relocation of DNB-related parameters, specified in TS 3/4.2.5, to the COLR.

As a result of the above changes, TS 6.8.1.6, Core Operating Limits Report, and associated TS Bases must be revised to be reflective of the above changes. Furthermore, editorial and administrative changes to TS 6.8.1.6, consistent with ITS, are proposed as well.

The proposed changes are either based on (1) NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications", dated October 3, 1988; (2) the NRC staff's acceptance of WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report;" and/or (3) the improved Standard Technical Specifications, NUREG-1431, Revision 2.

Relocation of cycle-specific parameters from TS to the COLR (a licensee-controlled document subject to the requirements of TS 6.8.1.6 and the provisions of 10 CFR 50.59) would afford North Atlantic flexibility to revise cycle-specific parameters, in accordance with NRC-approved methodologies, without the need for license amendment submittals. Specifically, TS 6.8.1.6c requires copies of the COLR to be submitted to the NRC for each reload cycle, including any mid-cycle revisions or supplements thereto. Thus resources, both North Atlantic and NRC, would be saved by minimizing and/or eliminating repetitive LAR submittals associated with revising cycle-specific parameters.

In accordance with 10 CFR 50.92, North Atlantic has concluded that the proposed change does 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 relocate cycle-specific parameters from TS to the COLR are administrative in nature and do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which it is operated. The proposed changes do not alter or prevent the ability of structures, systems or components to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Report.

The subject parameter limits will continue to be administratively controlled in accordance with Technical Specification 6.8.1.6. Specifically, TS 6.8.1.6.c requires the COLR to be submitted to the NRC each reload cycle, including any mid-cycle revisions or supplements.

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

2. **The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.**

The proposed changes do not alter the design assumptions, conditions, or configuration of the facility or the manner in which it is operated. There are no changes to the source term or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station Updated Final Safety Report (UFSAR). The proposed changes have no adverse impact on component or system interactions. The proposed changes will not degrade the ability of systems, structures or components important to safety to perform their safety function nor change the response of any system, structure or component important to safety as described in the Seabrook Station UFSAR. The proposed changes are administrative in nature and do not change the level of programmatic and procedural details that assure the operation of the facility in a safe manner. Since there are no changes to the design assumptions, parameters, conditions and configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new or different accident from any previously analyzed.

3. **The proposed changes do not involve a significant reduction in the margin of safety**

There is no adverse impact on equipment design or operation and there are no changes being made to Technical Specification cycle-specific parameter limits themselves that would adversely affect plant safety. The proposed changes are administrative in nature and imposes alternative procedural and programmatic controls on these parameter limits in accordance with the Commission's position established by Generic Letter 88-16 and the Safety Evaluation of WCAP-14483-A. Future changes to these limits will be submitted to the NRC in accordance with Technical Specification 6.8.1.6.

Therefore, relocation of the subject cycle-specific parameter limits and other proposed editorial changes, to be reflective of the relocated parameters, do not involve a significant reduction in the margin or safety provided in the existing specifications.

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 Staff review of License Amendment Request 01-10 and issuance of a license amendment by April 15, 2003, becoming effective immediately and implemented within 90 days thereafter. The requested issuance date is based on NRC average turnaround time for non-outage related LARs.

VI ENVIRONMENTAL IMPACT ASSESSMENT

North Atlantic has reviewed the proposed license amendment against the criteria of 10CFR51.22 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 off-site, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, North Atlantic concludes that the proposed changes meets the criteria delineated in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.