February 28, 2005

Mr. Mark E. Warner, Site Vice President
c/o James M. Peschel
Seabrook Station
FPL Energy Seabrook, LLC
PO Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1- ISSUANCE OF AMENDMENT RE: 5.2 PERCENT POWER UPRATE (TAC NO. MC2364)

Dear Mr. Warner:

The Commission has issued the enclosed Amendment No. 101 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No. 1 (SS). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated March 17, 2004, as supplemented by letters dated March 17 (proprietary information), April 1, May 26, September 13 (2 letters), October 12, October 28, December 3, December 28, 2004, January 28, February 9, and February 25, 2005.

The amendment revises the SS operating license and TSs to increase the licensed rated power by 5.2 percent from 3411 megawatts thermal to 3587 megawatts thermal.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Victor Nerses, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosures: 1. Amendment No. 101 to NPF-86
2. Safety Evaluation

cc w/encls: See next page

February 28, 2005

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Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosures: 1. Amendment No. 101 to NPF-86
2. Safety Evaluation

cc w/encls: See next page

Accession No.: ML050140453 Package No.: TSs:

OFFICE	PDI-2/PM	PDI-2/LA	SPLB/SC	SPLB/SC(A)	EMCB/SC	EMCB/SC	EMCB/SC	EEIB/SC	EEIB/SC	
NAME	VNerses	CRaynor	SWeerakkody	JTatum	SCoffin	TChan	LLund	EMarinos	RJenkins	
DATE	02/28/05	02/28/05	11/07/04	12/29/04	11/10/04	10/25/04	10/27/04	12/16/04	11/24/04	
OFFICE	IROB/SC	EMEB/SC	IPSB/TL	SRXB/SC	SPSB/SC	IROB/SC	OGC	PDI-2/SC	PDI/D	DLPM/D
NAME	DTrimble	KManoly	SKlementowicz	JUhle/DCoe	RDennig	TBoyce	TSmith	DRoberts	DRoberts for CHolden	TMarsh
DATE	02/10/05	12/07/04	01/31/05	01/11/05 02/28/05	02/02/05	02/10/05	02/25/05	02/28/05	02/28/05	02/28/05

OFFICIAL RECORD COPY

DATED: FEBRUARY 28, 2005

AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-86, SEABROOK STATION, UNIT NO. 1

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FPL ENERGY SEABROOK, LLC, ET AL.*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 101
License No. NPF-86

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by FPL Energy Seabrook, LLC, et al. (the licensee), dated March 17, 2004, as supplemented by letters dated March 17 (proprietary information), April 1, May 26, September 13 (2 letters), October 12, October 28, December 3, December 28, 2004, January 28, February 9, and February 25, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

---

*FPL Energy Seabrook, LLC (FPLE Seabrook), is authorized to act as agent for the: Hudson Light & Power Department, Massachusetts Municipal Wholesale Electric Company, and Taunton Municipal Light Plant and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-86 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 101, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Further, Facility Operating License No. NPF-86, will be amended to add license condition 2.K, to read as follows:

   K. <u>Inadvertent Actuation of the Emergency Core Cooling System (ECCS)</u>

   Prior to startup from refueling outage 11, FPL Energy Seabrook commits to either upgrade the controls for the pressurizer power operated relief valves to safety-grade status and confirm the safety-grade status and water-qualified capability of the pressurizer power operated relief valves, pressurizer power operated relief valve block valves and associated piping or to provide a reanalysis of the inadvertent safety injection event, using NRC-approved methodology, that concludes that the pressurizer does not become water-solid within the minimum allowable and verifiable time for operators to terminate the event.

3. This license amendment is effective as of its date of issuance and shall be implemented within 12 months of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Ledyard B. Marsh, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to Facility Operating License
and Technical Specifications

Date of Issuance: February 28, 2005

J. <u>Additional Conditions</u>

The Additional Conditions contained in Appendix C, as revised through Amendment No. 94, are hereby incorporated into this license. FPL Energy Seabrook, LLC, shall operate the facility in accordance with the Additional Conditions.

K. <u>Inadvertent Actuation of the Emergency Core Cooling System (ECCS)</u>

Prior to startup from refueling outage 11, FPL Energy Seabrook commits to either upgrade the controls for the pressurizer power operated relief valves (PORV) to safety-grade status and confirm the safety-grade status and water-qualified capability of the PORVs, PORV block valves and associated piping or to provide a reanalysis of the inadvertent safety injection event, using NRC approved methodologies, that concludes that the pressurizer does not become water solid within the minimum allowable time for operators to terminate the event.

3. This License is effective as of the date of issuance and shall expire at midnight on October 17, 2026.

FOR THE NUCLEAR REGULATORY COMMISSION

(Original signed by:
Thomas E. Murley)

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Appendix A - Technical Specifications (NUREG-1386)
2. Appendix B - Environmental Protection Plan
3. Appendix C - Additional Conditions

Date of Issuance: March 15, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 101

FACILITY OPERATING LICENSE NO. NPF-86

DOCKET NO. 50-443

Replace the following pages of the Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3	3
7	7

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
1-5	1-5
2-1	2-1
2-5	2-5
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
3/4 2-6b	3/4 2-6b
3/4 2-10	3/4 2-10
3/4 3-26	3/4 3-26
3/4 2-27	3/4 2-27
3/4 7-2	3/4 7-2
6-18A	6-18A
6-18B	6-18B
6-18C	6-18C
6-18D	6-18D
B 3/4 2-3	B 3/4 2-3
B 3/4 7-1	B 3/4 7-1

Seabrook Station, Unit No. 1

Safety Evaluation for Amendment No. 101

Dated: February 28, 2005

Regarding 5.2% Power Uprate

TABLE OF CONTENTS

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-86

FPL ENERGY SEABROOK, LLC

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By application dated March 17, 2004 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML040860307), as supplemented on March 17, April 1, May 26, September 13 (2 letters), October 12, October 28, December 3, December 28, 2004, January 28, February 9, and February 25, 2005 (ADAMS Accession Nos. ML040850074, ML041050175, ML041560339, ML042660100, ML042660272, ML042890281, ML043060466, ML043440030, ML043650401, ML050320112, and ML050450414, respectively), FPL Energy Seabrook, LLC (FPLE or licensee) submitted a request for changes to the Seabrook Station, Unit No. 1 (SS) Technical Specifications (TSs). The proposed amendment would increase the licensed reactor core power level by 5.2% from 3411 megawatts thermal (MWt) to 3587 MWt. Based on its review of this application, the Nuclear Regulatory Commission (NRC or the Commission) staff categorized the application as a stretch power uprate (SPU). The modifications required to achieve the 5.2% SPU at SS are planned for the refueling outage 10.

Specifically, the following are the proposed changes:

1. Facility Operating License NPF-86, paragraph 2.C(1), “Maximum Power Level” would change to authorize operation at reactor core power levels not in excess of 3587 MWt.

2. A new license condition 2.K “Inadvertent Actuation of the Emergency Core Cooling System,” is being added.

3. Rated Thermal Power (RTP) value in TS Section 1.0, paragraph 1.28, would be changed from 3411 MWt to 3587 MWt.

4. REACTOR CORE SAFETY LIMITS in TS Section 2.1.1, would be changed from 1.17 to 1.14 and would delete the WRB-1 and WRB-2 correlations.

5. Changes in Trip Setpoints in Table 2.2-1 (REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS):

    - Functional Unit 13., Steam Generator Water Level Low - Low. The TOTAL ALLOWANCE (TA), Z and SENSOR ERROR (S) would change from 14.0, 12.53, and 0.55, respectively to N.A. The TRIP SETPOINT would change from

$14.0% to $20.0% of narrow range instrument span . The ALLOWABLE VALUE would change from $12.6% to $19.5% of narrow range instrument span.

6. Various table notations on TS pages 2-7, 2-8, 2-9 and 2-10 would change to correct typographical errors .

7. The POWER DISTRIBUTION LIMITS (PDL), HEAT FLUX HOT CHANNEL FACTOR, SURVEILLANCE REQUIREMENTS in TS Section 4.2.2.2.g.1), Lower Core Region would change from 0 to 15%, inclusive to 0 to 10%, inclusive and TS Section 4.2.2.2.g.2), Upper Core Region would change from 85 to 100%, inclusive to 90 to 100%, inclusive.

8. The PDL DEPARTURE FROM NUCLEATE BOILING PARAMETERS, LIMITING CONDITION FOR OPERATION in TS Section 3.2.5.c.1, Reactor Coolant System (RCS) Flow would change from $382,800 gpm to $374,400 gpm, and TS Section 3.2.5.c.2, Reactor Coolant System Flow would change from $392,800 gpm to $383,800 gpm.

9. Changes in allowable values (AVs) in Table 3.3-4, (Engineered Safety Features Actuation System (ESFAS) Instrumentation Trip Setpoints):

- FUNCTIONAL UNIT 5.b, Steam Generator Water Level -- High-High (P-14), The TOTAL ALLOWANCE (TA), Z AND SENSOR ERROR (S) would change from 4.0, 2.24 and .55, respectively to N. A. The TRIP SETPOINT would change from 86.0% to 90.8% of narrow range instrument span. The ALLOWABLE VALUE would change from 87.7% to 91.3% of narrow range instrument span.

- FUNCTIONAL UNIT 6.a, Steam Generator Water Level -- Hi-Hi (P-14). The TOTAL ALLOWANCE (TA), Z AND SENSOR ERROR (S) would change from 4.0, 2.24 and .55, respectively to N. A. The TRIP SETPOINT would change from 86.8% to 90.8% of narrow range instrument span. The ALLOWABLE VALUE would change from 87.7% to 91.3% of narrow range instrument span.

- FUNCTIONAL UNIT 7.c, Steam Generator Water Level–Low-Low. The TOTAL ALLOWANCE (TA), Z AND SENSOR ERROR (S) would change from 14.0, 12.53 and .55, respectively to N. A. The TRIP SETPOINT would change from 14.0% to 20.0% of narrow range instrument span. The ALLOWABLE VALUE would change from 12.6% to 19.5% of narrow range instrument span.

10. Changes in Table 3.7-1, MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR-LOOP OPERATION:

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT with one inoperable safety valve on any operating steam generator, the percent of rated thermal power would change from 66 to 60; with two inoperable safety valves on any operating steam generator, the percent of rated thermal power would change from 47 to 42; with three inoperable safety valves on any operating steam generator, the percent of rated thermal power would change from 28 to 25.

11. Changes to the Administrative Controls Section 6.8.1.6.b.1:

    - Add the reference WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 thorough 5 (Revision 1,) “Code Qualification Document for Best Estimate LOCA [loss-of-coolant accident] Analysis” March 1998

12. Changes to the Administrative Controls Section 6.8.1.6.b.5:

    - Change WCAP-14565-P to WCAP-14565-P-A and the date April 1997 to October 1999. Delete in their entirety the words beginning with “Letter from T.H. Essig........” and ending with “.....January 1999". Change WCAP-15025-P to WCAP-15025-P-A and the date February 1998 to April 1999.

13. Changes to the Administrative Controls Section 6.8.1.6.b.6:

    - Add the reference: WCAP-8745-P-A, “Design Basis For the Thermal Overpower T and Thermal Overtemperature ΔT Trip Functions,” September 1986.

14. Changes to the Administrative Controls Section 6.8.1.6.b.7:

    - Delete in their entirety the words beginning with “WCAP-14551-P,.......” and ending with “...June 1998.”

15. Changes to the Administrative Controls Section 6.8.1.6.b.14, page 6-18D.

    - Delete in their entirety the words beginning with “WCAP-8385-P,.......” and ending with “...September 1974.”

The licensee stated that the appropriate TS Bases will be revised associated with certain of above changes.

Attachment 1 to the March 17, 2004, submittal contains the technical assessment and safety analysis of the proposed power uprate. The licensee considers the uprate to be an SPU because, consistent with the guidance in NRC Review Standard RS-001, “Review Standard for Extended Power Uprates,” the power increase is less than 7% and there are no major plant modifications. Nevertheless, the licensee used the guidance in RS-001 to develop their application.

The supplements dated March 17 (proprietary information), April 1, May 26, September 13 (2 letters), October 12, October 28, December 3, December 28, 2004, January 28, February 9, and February 25, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on June 22, 2004 (69 FR 34701).

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified core thermal power. SS was initially licensed to operate at a maximum of 3411 MWt. However, various systems and components were designed to accommodate the conditions associated with a power level of 3587 MWt. The proposed SPU of 5.2% will allow the licensed rated power to be increased from the current value of 3411 MWt to 3587 MWt.

3.0 REGULATORY AND TECHNICAL EVALUATION

In several places in this safety evaluation (SE), the NRC staff refers to NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants LWR [light-water reactor] Edition," as guidance used during the review. The NRC staff notes that the SRP was used for general technical guidance. The licensee's March 17, 2004, application, as supplemented, was reviewed for compliance with the SS licensing basis.

3.1 Instrumentation and Controls (I&Cs)

3.1.1 Regulatory Evaluation

I&C systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (control rods), (3) to initiate the engineered safety feature (ESF) systems and essential auxiliary supporting systems, and (4) to achieve and maintain a safe shutdown condition of the plant. Diverse I&C systems and equipment are provided for the express purpose of protecting against potential common-mode failures of I&C protection systems. The NRC staff conducted a review of the reactor trip system, ESFAS, safe shutdown systems, control systems, and diverse I&C systems for the proposed power uprate to ensure that the systems and any changes required for the proposed power uprate are adequately designed such that the systems continue to meet their safety functions. The NRC staff's review was also conducted to ensure that failures of the systems do not affect safety functions.

Nuclear power plants are licensed to operate at a specified core thermal power. The measurement uncertainties are considered at that power level to avoid exceeding the power levels assumed in the design basis transient and accident analysis. Furthermore, the safety trip setpoints (TSPs) are calculated to ensure that sufficient allowance exists between the TSP and the safety limit (SL) to account for instrument uncertainties. The Commission's regulatory requirements related to this review can be found in Title 10 of the *Code of Federal Regulations* (10 CFR) as follows:

- Section 50.36(c)(1)(ii)(A) of 10 CFR requires that, where a limiting safety system setting (LSSS) is specified for a variable on which a SL has been placed, the setting be chosen so that automatic protective action will correct the abnormal situation anticipated without exceeding a SL. LSSS are settings for automatic protective devices related to variables having significant safety functions. Setpoints found to exceed TSs limits are considered a malfunction of an automatic safety system. Such an occurrence could challenge the integrity of the reactor core, reactor coolant pressure boundary (RCPB), containment, and associated systems.

- Regulatory Guide (RG) 1.105, Revision 3, "Setpoint for Safety-Related Instrumentation," is used to evaluate the conformance with 10 CFR 50.36.

- The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and General Design Criteria (GDCs) 1, 4, 13, 19, 20, 21, 22, 23, and 24. Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

3.1.2 Technical Evaluation

3.1.2.1 Suitability of Existing Instruments

The SS plant protection systems are the reactor protection system (RPS) and the ESFAS. The RPS is designed to trip the reactor by de-energizing the control element drive mechanisms whenever any monitored condition reaches a TSP. For each measured variable, the RPS uses a 2-out-of-4 channel logic arrangement, with each channel electrically and physically separated to ensure no loss of functionality with a single failure. The RPS is designed to automatically keeps the reactor operating within a safe region by shutting down the reactor whenever the limits of the region are approached. The reactor trip system keeps surveillance on process variables and whenever a direct process or calculated variable exceeds a setpoint, the reactor will be shut down to protect against either gross damage to fuel cladding or loss of system integrity which could lead to release of radioactive fission products into the containment. The ESFAS is designed to initiate safety features whenever any monitored condition reaches a TSP. The ESFAS also uses a 2-out-of-4 logic arrangement, with each channel electrically and physically separated to ensure no loss of functionality with a single failure. The ESFAS is designed to mitigate the consequences of certain design basis accidents (DBAs), particularly by protecting the integrity of the containment building.

The RPS is discussed in Section 7.2 of the SS updated final safety analyses report (UFSAR). The ESFAS is discussed in UFSAR Section 7.3. Systems required for safe shutdown are described in UFSAR Section 7.4. Instrumentation required to monitor, control, and provide interlocks for these systems is described in UFSAR Sections 7.2, 7.3, 7.6, and 7.7. The anticipated transient without scram (ATWS) mitigation system is described in UFSAR Section 7.6.

The change resulting from the SS SPU is that the full power T_{avg} window increases from 571.0 °F to 589.1 °F. To address this change, the pressurizer level control program was revised. There were no other nuclear steam supply system (NSSS) control system setpoint changes required for the SPU. Therefore, the impact of this change on the NSSS control systems for the SPU was evaluated. The RPS and ESFAS setpoints for low-low steam generator (SG) level trip and high-high SG level ESFAS were revised. The revised setpoints were evaluated to determine the impact on the plant operating margin.

The licensee stated that the methodology used to calculate the values for these constants is consistent with the past practice and NRC-approved methods. In Section 3.2.2 of this SE, the NRC staff found this methodology acceptable because of previous NRC approval.

For SG level, the transmitters are Rosemount Model 1154DP4RA and the process racks are Westinghouse (W) 7300 racks. The narrow range span is defined as the distance between the lower instrument tap and the upper instrument tap. The distance is 127.8 inches. The transmitter range is 150 inches of water column and the calibrated span is 85.72 inches of water column. The process rack range and span is 0 - 10 volts dc (direct current). The physical unit of measure for this function is level. Based on the tap-to-tap distance of 127.8 inches, the previous TS low-level setpoint of 14% narrow range span is 17.9 inches above the lower tap and the previous TS high-level setpoint of 86% narrow range span is 109.9 inches above the lower tap. The setpoint change for the SPU is 20% of narrow range span, which is 25.6 inches above the lower tap, and the high-level setpoint change is 90.8% of narrow range span, which is 116 inches above the lower tap.

The SPU does not change the safety functions or design requirements such as separation, redundancy, or diversity of the I&Cs as described in UFSAR Sections 7.2, 7.3, 7.4, 7.5, 7.6, and 7.7. The I&C-related changes resulting from the SPU are consistent with the licensing basis and comply with acceptance criteria related to 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), 10 CFR Part 50 Appendix A, and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Therefore, the staff finds that the existing instruments at SS are suitable for SPU operations.

3.1.2.2 Instrument Setpoints Methodology

3.1.2.2.1 Generic Concern Regarding Method 3

During recent reviews of proposed license amendments associated with changes to the TS LSSS, the staff has identified a concern regarding the method used by some licensees to determine the TS AVs. AVs are used in the TS as LSSS, to provide acceptance criteria for determination of instrument channel operability during periodic surveillance testing. The staff's concern relates to one of the three methods for determining the AV as described in Instrument Society of America (ISA) Recommended Practice ISA-RP67.04-1994, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation."

Paragraph (c)(1)(ii)(A) of 10 CFR 50.36, "Technical Specifications," states, in part, that where an LSSS is specified for a variable on which a SL has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. The analytical limit (AL) is the limit on the process variable at which the instrument loop protective action occurs as assumed in the plant's safety analysis. Protective action at the AL ensures that the SL is not exceeded. The AL, however, does not account for uncertainties associated with the instrument loop. The instrument loop uncertainty is accounted for during calculation of an instrument loop's TSP.

Method 3 in ISA-RP67.04-1994 calculates the TSP by subtracting the instrument loop uncertainty, also known as total loop uncertainty, from the AL. The AV is then determined by adding the uncertainties measured during periodic surveillance testing (e.g., drift, calibration uncertainties, instrument accuracy) to the TSP.

The staff concern is that an AV determined by Method 3 does not ensure adequate margin between the AV and the AL to account for uncertainties which are not addressed in periodic surveillance testing. Some examples of the way Method 3 has been used omitted uncertainties associated with instruments excluded from the tests, and uncertainties due to the effects of normal environmental variation. These additional uncertainties could result in sufficient error in the channel setpoint to cause the channel trip to occur only after the process variable has already exceeded the AL, even though the measured value of the setpoint is within the Method 3 AV. This could result in operation under conditions not addressed in the plant safety analyses, with attendant risk of violation of associated SLs. A determination that the measured as-found channel setpoint is within the Method 3 AV could give a false indication that the SLs were adequately protected. In such a case no corrective action would be initiated and the potential violation of the associated SLs would go unnoticed. This concern was discussed in a meeting with the Nuclear Energy Institute (NEI) on October 8, 2003, as described in the NRC's meeting summary dated October 28, 2003 (ADAMS Accession No. ML033030193). NEI provided its position on this issue in a letter dated December 5, 2003 (ADAMS Accession No. ML033450410). By letter dated December 17, 2004, NEI submitted their response to the staff's letter dated June 17, 2004. The staff is in the process of reviewing this document and has planned a meeting for March 11, 2005, with NEI to present its response to the NEI letter. The staff is currently working towards resolution of this generic concern.

3.1.2.2.2 Acceptability of the Proposed Changes

The licensee stated, on page 7 of Enclosure 1 to their October 12, 2004, request for additional information (RAI) response, that the AVs determined for the SPU were not calculated using Method 3 of ISA-S67.04.02. The licensee stated that the method used in its channel uncertainty and setpoint determination, was as described in WCAP-10991,"Westinghouse Setpoint Methodology for Protection Systems" (ADAMS Accession No. 9710200073). This methodology statistically combines the individual uncertainties using the square root of the sum of the squares method in accordance with ISA-S67.04. The methodology to determine operability, also described in WCAP-10991, is a performance based AV. The AV is satisfied by verification that the channel "as found" and "as left" conditions about the nominal TSP are within the rack calibration accuracy. These criteria are controlled by the TSs and implemented by plant procedures. In the SS TS Section 2.2.1 Parts a and b, pages 2-3, and TS 3.3.2 Parts a and b, pages 3-14, the requirement is to return the instrumentation to the nominal TSP. Because the AV is based on the rack calibration accuracy, the TS requirement is to return the channel to within the calibration accuracy. Therefore, the TS requirement returning the channel to within the calibration accuracy resolves the staff's concern with Method 3. Since the operability determination used only the TSP and AV, the values for Total Allowance, Z, and Sensor Error are changed to N.A.

Therefore, the licensee's setpoint methodology, which has been discussed in WCAP-10991 and has previously been reviewed and approved in a staff SE dated May 26, 1998 (ADAMS Accession No. ML011790264), is acceptable.

3.1.2.3 I&C-Related TS Changes Related to the Power Uprate

3.1.2.3.1 Reactor Trip System Instrumentation TSPs and AVs

TS Table 2.1-1, Reactor Trip System Instrumentation Trip Setpoints, Functional Unit 13, Steam Generator Water Level Low-Low is revised as follows:

A) The Trip Setpoint is changed from >14.0% to >20.0% of narrow range instrument span.
B) The AV is changed from >12.6% to >19.5% of narrow range instrument span.
C) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

The basic function of the reactor protection circuits associated with low-low SG water level is to preserve the SG heat sink for removal of long-term residual heat. In the event of a complete loss of feedwater (FW), the reactor would be tripped on low-low SG water level and emergency feedwater pumps (FWPs) would provide FW to maintain residual heat removal (RHR) after trip. This reactor trip acts before the SGs are dry. Therefore, a low-low SG water level reactor trip circuit is provided for each SG to ensure that sufficient initial thermal capacity is available in the SG at the start of the transient. With the added power after the power uprate, the full power T_{avg} window is from 571.0 °F to 589.1 °F.

Enclosure 2 of the October 12, 2004 RAI response contains detailed process measurement errors and instrument uncertainties, as well as the calculated TSP and AV. The staff finds that the TSP calculation meets the requirements of 10 CFR 50.36 and RG 1.105 and, therefore, is acceptable.

To ensure acceptable plant response and adequate plant operating margins at the SPU power conditions, the American Nuclear Society (ANS) Condition I transients were analyzed using the LOFTRAN computer code and the following conditions:

- 50% load rejection
- 5%/minute ramp load changes
- 10% step load changes
- Turbine trip without reactor trip from P-9 setpoint

The major analysis assumptions are as follows.

- Analyses for both low (571.0 °F) and high (589.1 °F) RCS full power T_{avg} and 0% and 10% average SG tube plugging conditions
- Credit for the NSSS control systems (rod control, steam dump control, and pressurizer pressure control) in automatic mode of control
- No credit for pressurizer and SG safety valves
- Best estimate fuel reactivity feedbacks [moderator temperature coefficient (MTC), Doppler power defect and control rod worth] at beginning of life (BOL) core conditions, which are limiting for the margin to trip assessment
- No credit for operator action

The licensee stated that analysis showed that the low-low SG level reactor TSP is adequate. The staff's evaluation of these transients and assumptions are included in Section 3.2 of this SE. The staff finds that the TSP calculation meets the requirements of 10 CFR 50.36 and RG 1.105 and, therefore, is acceptable.

3.1.2.3.2 ESFAS Instrumentation TSPs and AVs

In addition to the requirements for a reactor trip for anticipated abnormal transients, the ESFAS is provided with instrumentation and controls to sense accident situations and initiate the operation of necessary ESFs. The occurrence of a limiting fault, such as a LOCA or a steamline break, requires a reactor trip plus actuation of one or more of the ESFs to prevent or mitigate damage to the core and RCS components, and ensure containment integrity.

The specific functions that rely on the ESFAS for initiation and which are being modified as a result of this power uprate are the turbine trip, the start of motor-driven and turbine-driven emergency FWPs, and main FW line isolation (as required to prevent or mitigate the effect of excessive cooldown). The following TS changes were requested by the licensee:

(1) TS Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints, Functional Unit 5b, Turbine Trip, Steam Generator Water Level High-High (P14) is revised as follows:

A) The Trip Setpoint is changed from #86.0% to #90.8% of narrow range instrument span.
B) The Allowable Value is changed from #87.7% to #91.3% of narrow range instrument span.
C) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

(2) TS Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints, Functional Unit 6a, Feedwater Isolation, Steam Generator Water Level Hi-Hi (P14) is revised as follows:

A) The Trip Setpoint is changed from #86.0% to #90.8% of narrow range instrument span.
B) The Allowable Value is changed from #87.7% to #91.3% of narrow range instrument span.
C) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

(3) TS Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints, Functional Unit 7c, Emergency Feedwater, Steam Generator Water Level Low-Low, Start Motor-Driven Pump and Start Turbine-Driven Pump is revised as follows:

A) The Trip Setpoint is changed from >14.0% to >20.0% of narrow range instrument span.
B) The Allowable Value is changed from >12.6% to >19.5% of narrow range instrument span.
C) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

The same analysis as used in RPS setpoint determination, discussed in Section 3.1.2.3.1 above, was used to show that the low-low SG level ESF TSP and high-high SG water level ESF setpoints were adequate. Enclosure 2 of the October 12, 2004 RAI response contains detailed process measurement errors and instrument uncertainties, as well as the calculated TSP and AV. The staff finds that the TSP calculation meets the requirements of 10 CFR 50.36 and RG 1.105 and, therefore, is acceptable.

3.1.2.3.3 Summary

The NRC staff has reviewed the licensee's application related to the effects of the proposed 5.2% SPU on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. The NRC staff concludes that the licensee has adequately addressed the effects of the proposed 5.2% SPU on these systems and that the changes that are necessary to achieve the proposed 5.2% SPU are consistent with the plant's design basis. The NRC staff further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55(a)(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Therefore, the NRC staff finds the licensee's proposed 5.2% SPU acceptable with respect to I&Cs.

3.2 Reactor Systems

3.2.1 Regulatory Evaluation

The staff reviewed the licensee's evaluations and analyses supporting a 3587 MWt SPU. Although the licensee has not requested a power uprate that is high enough to qualify as an extended power uprate, the staff performed its review using the Review Standard for Extended Power Uprates [Reference 6] as a guideline. The staff's review included the following areas: nuclear and fuel design; thermal-hydraulic design; systems evaluations; and LOCA and non-LOCA transient and accident analyses. A discussion of the computer codes and methodologies used in the SPU application can be found in Section 3.2.13 of this SE. Each of the review areas is evaluated separately according to its applicable regulatory requirements and acceptance criteria, the Standard Review Plan, NUREG-0800 [Reference 7], and the results of the licensee's analyses or evaluations

3.2.2 NSSS Parameters

The NSSS design parameters provide the RCS and secondary system conditions for use in NSSS analyses and evaluations. The licensee provided a list of key plant parameters corresponding to the proposed SPU level of 3587 MWt in Table 2.3-1 [Reference 1]. The major parameters include reactor power level, NSSS power level, thermal design flow (TDF), reactor coolant pressure (RCP) and temperatures, SG pressure, steam temperature and steam flow rate. The major changes of these design parameters from the current values include increased core power level, decrease in the core inlet temperature, lower maximum steam pressure, lower maximum steam temperature, and a higher steam flow rate. In response to the staff's RAI, the licensee provided a new table listing thermal design parameter values and safety analyses parameters values. The licensee used a range of conditions for the vessel average temperature (T_{avg}), the SG tube plugging level, and a range of FW temperature to generate the design operating parameters. The vessel T_{avg} range was between 571.0 EF and 589.1 EF, the SG tube plugging level can vary from 0% to 10.0%, and FW temperature range was between 390 EF to 452.4 EF. The licensee also considered 2% initial conditions uncertainties in its safety analyses. These parameters are used in the licensee's safety analyses performed to support its proposed power uprate. The analyses demonstrated the plant's acceptable margin to safety analysis limits (SALs) and provides an operational flexibility. The NRC staff evaluated these changes and found them to adequately represent the plant operating conditions at the proposed core power level of 3587 MWt. Therefore, the NRC staff finds the NSSS design parameters acceptable.

3.2.3 RCS

The changes in NSSS design parameters that impact the RCS design basis functions include the increase in core power and an allowable range for RCS $T_{avg.}$. The minimum measured flow (MMF) stated in the TS reduced from 392,800 gpm to 383,800 gpm. The TDF of 95,700 gpm per loop was reduced to 93,600 per loop due to a 10% SG tube plugging consideration. The steady-state RCS pressure (2235 psig) and no-load RCS temperature (557 EF) have not changed. The RCS temperature associated with the proposed SPU remains within the bounds of the original design temperature of 650 EF for the RCS and 680 EF for the pressurizer. Sufficient core cooling under power uprate conditions is verified by various plant transient and safety analyses. The NRC staff finds that the changes of RCS operating parameters associated with the power uprate are acceptable based on the results of the safety analyses addressed in Section 3.2.13 below.

3.2.4 Safety Injection System (SIS)

The licensee verified the adequacy of the SIS during the injection and sump recirculation phases following a LOCA in the LOCA analysis performed assuming a core power level of 3659 MWt. For the non-LOCA events, the performance of SIS was also verified by various safety analyses performed in support of the proposed power uprate. The licensee concluded that no system modifications are required to support the proposed SPU. The staff agrees with the licensee's assessment based on the acceptable results of the safety analyses addressed in Section 3.2.13, below.

3.2.5 RHR System

Operation at a higher power level increases the amount of decay heat being generated in the core, which results in a higher heat load to the RHR system for plant cooldown and also during refueling. The increased heat loads will be transferred to the primary component cooling water (CCW) system and ultimately to the service water system. Normal cooldown is accomplished with two residual heat exchangers and two primary CCW heat exchangers in service. The cooldown evaluation performed with two trains from 350 EF to 125 EF is within the cooldown time of 20 hours at SPU operating conditions. When one train of RHR is in operation, the reactor coolant is reduced from 350 EF to 200 EF within 33.5 hours, which is within the current TS limit of 37 hours. For the cases ran under SPU conditions, the licensee confirmed that the RHR cooldown capacity meets GDC-4, 5, and 34. Based on this evaluation, the licensee concluded that system modifications are not required to accommodate the SPU. The staff reviewed the licensee's evaluation and agrees with the licensee's assessment.

3.2.6 NSSS Transients

In its power uprate application, the licensee evaluated the NSSS design transients to account for any power uprate impacts. The NSSS design transients are traditionally developed for stress analyses of the various NSSS components using conservative assumptions. The licensee provided a tabulation comparing the plant operating conditions at the current power rating and the proposed NSSS power level of 3587 MWt. The licensee compared the design parameters used in the existing design transients and for the SPU parameters and concluded that the existing design transients remain bounding and applicable for the SPU. Even though the existing design transients bound the SPU program, all of the design transients were re-analyzed based on the SPU program design parameters to show that the regulatory requirements are still met.

3.2.7 Fuel System Design Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, and reactivity control rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents.

The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance; (2) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs; (3) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (4) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

The staff reviewed the licensee's analyses for the fuel design under SPU conditions. Rod internal pressure is considered a driving force for fuel system damage that could contribute to the loss of dimensional stability and cladding integrity. The SPU core power level results in higher fuel operating temperatures, which could increase the potential for increased fission gas release. Although an analysis of the representative rod power histories indicated that the higher duty fuel rods have increased fission gas release from the pellet resulting in higher rod internal pressures, the results meet the acceptance criteria. Therefore, the fuel system is not predicted to incur damage due to excessive fuel rod internal pressure at SPU conditions.

Based on the results of the applicant's analysis, the staff finds that the rod internal pressure analysis is acceptable for SS under SPU conditions.

SRP Section 4.2 identifies cladding oxidation buildup as a potential damage mechanism for fuel designs. The SRP further states that the effect of cladding oxidation needs to be addressed in safety and design analyses such as in the thermal and mechanical analysis.

The calculated fuel clad temperature (metal-oxide interface temperature) must be less than the license limit temperature for ZIRLO clad fuel during steady state operation. For ANS Condition II events, the calculated fuel clad temperature must not exceed the license limit for ZIRLO clad fuel. The hydrogen pickup level in the fuel clad must be less than or equal to the license limit at the end of fuel operation.

The SPU conditions entail an increase in operating temperatures for the fuel clad due to the higher fuel rod average power rating. Since the corrosion process is strongly influenced by fuel clad temperature, the SPU will make it harder to satisfy these criteria. Analysis of the representative rod power histories indicated that the corrosion design criteria are met for the higher duty fuel rods at the SPU core conditions. Based on the licensee's analysis results, the staff concludes that the impact of corrosion on the thermal and mechanical performance will be minimal for SS under SPU conditions.

The fuel rod strain fatigue capability could be impacted by SPU conditions of higher operating temperature and longer cycle length. The licensee re-analyzed the strain fatigue capability under SPU conditions, and showed that the fuel system design maintained its strain fatigue capability. Based on the results of the licensee's analysis, the staff concludes that the strain fatigue capability is acceptable for SS under SPU conditions.

SRP Section 4.2 states that the stress and strain limits in fuel designs should not be exceeded for normal operation and AOOs. During SPU conditions, the fuel system could experience high power duty loading, thereby exceeding the stress and strain limits for certain AOOs. The licensee re-examined the fuel system loading using the PAD 4.0 code [References 12 and 13] to analyze the stress and strain conditions. The results showed that the stress and strain limits were not exceeded for SPU conditions. Based on the approved analyses, the staff concludes that the fuel system design meets the stress and strain limits for SS under SPU conditions.

The staff reviewed the licensee's analyses related to the effects of the proposed SPU on the fuel system design. The staff concludes that the licensee has adequately accounted for the effects of the proposed SPU on the fuel system and demonstrated that: the fuel system will not be damaged as a result of normal operation and AOOs; the fuel system damage will never be so severe as to prevent control rod insertion when it is required; the number of fuel rod failures will not be underestimated for postulated accidents; and coolability will always be maintained. Based on this, the staff concludes that the fuel system and associated analyses will continue to meet the regulatory requirements as noted above following implementation of the proposed SPU. Therefore, the staff finds the proposed SPU acceptable with respect to the fuel system design.

3.2.8 Nuclear Design Evaluation

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NRC staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity; (3) GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed; (4) GDC-13, insofar as it requires that I&Cs be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges; (5) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to ensure that acceptable fuel design limits are not exceeded as a result of AOOs, and to automatically initiate operation of systems and components important to safety under accident conditions; (6) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (7) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to ensure the capability to cool the core is maintained; and (9) GDC-28, insofar as it requires that the reactivity control systems be designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals (RVIs) so as to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

The nuclear design evaluation was performed to account for the SPU core power level of 3659 MWt, using approved nuclear design analytical models and methods [References 14, 15 and 16] to describe the neutronic behavior of the SPU core and assess the impact of the SPU on key nuclear design parameters.

The licensee states that it did not find it necessary to modify any of the W nuclear design philosophies or methods to evaluate the nuclear design under SPU conditions. The nuclear design evaluation addresses key nuclear design-related safety parameters that are input to the safety analyses [Reference 14], such as power distributions, peaking factors, control rod worths, boron concentrations, and reactivity coefficients. The methodology calls for the evaluation of these key safety parameters for each reload cycle. If any of the parameters fall outside the bounds assumed in the reference safety analysis, then the affected transients are re-evaluated.

The licensee built four cycles of core models that incorporated the SPU level. Conceptual loading patterns were developed and used to confirm the continued validity of existing key safety parameters, or to define revised parameter limits where needed. The loading patterns were also used to calculate representative power distributions for confirming the acceptability of transient statepoint conditions, and fuel rod performance parameters, as well as establishing necessary power distribution inputs for the best estimate large-break LOCA (LBLOCA) analysis.

Westinghouse has determined that, under certain SPU conditions, there could be peak steady-state and transient F_Q values in the bottom 15% of the core. The axial zone is currently excluded from the F_Q surveillance of SS TS Surveillance Requirement (SR) 4.2.2.2.g. As a result, the F_Q surveillance exclusion zone is changed from 15% to 10%.

The staff reviewed the licensee's analysis related to the nuclear design at the SPU of 3659 MWt. The licensee evaluated key safety parameters for the SPU core conditions, and calculated representative power distributions for use in confirming fuel design limits at nominal and transient statepoint power conditions and for establishing normal operating power distribution ranges for the best estimate LOCA analysis. Nuclear design criteria were met for the loading patterns generated at the SPU conditions. The licensee stated that each specific reload cycle core design will be such that it meets the nuclear design criteria at the SPU conditions. The staff agrees with the licensee's approach and results.

The staff concludes that the licensee adequately accounted for the effects of the proposed SPU on the nuclear design and has demonstrated that the fuel design limits will not be exceeded, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the regulatory requirements stated above following implementation of the proposed SPU. Therefore, the staff finds the proposed uprate acceptable with respect to the nuclear design.

3.2.9 Thermal and Hydraulic Design Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods; (2) is equivalent to or a justified extrapolation from proven designs; (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs; and (4) is not susceptible to thermal-hydraulic instability. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and (2) GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed. Specific review criteria are contained in SRP Section 4.4.

The staff reviewed the thermal and hydraulics safety analysis and evaluations performed to support operation of the SS core consisting of 17x17 robust fuel assemblies with intermediate flow mixer grids [Reference 19] at the analyzed SPU core power level of 3659 MWt.

The thermal and hydraulic, departure from nucleate boiling (DNB), safety analyses performed for the SPU, were based upon the Revised Thermal Design Procedure (RTDP) methodology [Reference 17] and the WRB-2M DNB correlation [Reference 18]. The RCS methodology applies statistical calculations to obtain DNB sensitivity factors for uncertainties (random portions only) in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters. The Standard Thermal Design Procedure (STDP) methodology and the WRB-2 [Reference 19] or W-3 DNB correlation were used when RCS and WRB-2M were not applicable. The analysis results indicate that the 95/95 DNB design basis is met for operation and AOOs under SPU conditions.

The W version of the VIPRE-01 Code was used to perform NDBR calculations with the WRB-2, WRB-2M and the W-3 DNB correlations. The application of VIPRE for these SPU analyses was within the conditions specified in the staff's Safety Evaluation Report (SER) contained in WCAP-14565-P-A [Reference 20].

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed SPU on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed SPU on the thermal and hydraulic design and demonstrated that the design: (1) has been accomplished using acceptable analytical methods; (2) is equivalent to proven designs; (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs; and (4) is not susceptible to thermal-hydraulic instability. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDCs 10 and 12 following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to thermal and hydraulic design.

3.2.10 Functional Design of Control Rod Drive System (CRDS)

The NRC staff's review covered the functional performance of the CRDS to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-23, insofar as it requires that the protection system be designed to fail into a safe state; (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (4) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (5) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (6) GDC-28, insofar as it requires that the reactivity control systems be designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other RVIs so as to significantly impair the capability to cool the core; and (7) GDC-29, insofar as it requires that the protection and reactivity control systems be designed to ensure an extremely high probability of accomplishing their safety functions in event of AOOs. Specific review criteria are contained in SRP Section 4.6.

The SS control rod drive mechanisms (CRDMs) were originally designed and analyzed to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) [Reference 21]. The input parameters that were used to perform the SPU analyses and evaluations include the SPU design operating parameters, NSSS design transients, and the current design basis evaluations for the CRDMs. There were no other changes to the pressure or thermal design parameters considered for the SPU. Seismic analyses and non-pressure boundary component evaluations are unaffected by the SPU.

The CRDMs are affected by RCP, reactor vessel (RV) head temperature, and cold leg NSSS design transients. The SPU does not change the RCP. Therefore, evaluations of the CRDMs remain valid for the SPU RCP condition.

The highest RV outlet temperature determined for the SPU is 621.4 EF. Since most of the previous analyses used material AVs based on the design temperature of 650 EF, the proposed temperature is enveloped by the previous analyses. The only evaluations that were not bounded were those associated with the changes in NSSS design transients that were not enveloped by the current analyses. Ratios of the new transients to the old transients were used (very small change, less than 5%) to multiply the existing stress evaluation results. After this was performed, it was shown that the component stresses were within the allowable limits of the ASME Code. The staff agrees with the licensee's approach and results.

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed SPU on the functional design of the CRDS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed SPU on the system and demonstrated that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed SPU. Based on this, the NRC staff concludes that the CRDS will continue to meet the requirements of GDCs 4, 23, 25, 26, 27, 28, and 29 following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to the functional design of the CRDS.

3.2.11 Reactor Pressure Vessel (RPV) Neutron Fluence Evaluation

The licensee conducted a reevaluation of the vessel fast neutron fluence (E > 1.0 MeV) to the end of the current license. The methodology adheres to the guidance in RG 1.190 [Reference 37]. With all other factors being the same, the neutron sources were conservatively estimated. The previous nine cycles were derived from the SS cycle loading reports but the current and future cycles were estimated on the basis of the most conservative loading projections.

The analytical predictions were then compared to the results of the SS dosimetry and established that the results were in excellent agreement, i.e., well within the 20% guidance in RG 1.190. The staff finds the fluence results acceptable, because they are derived using a methodology in agreement with RG 1.190, and in addition are confirmed by the plant specific dosimetry.

The fluence results impact the pressure-temperature (PT) limit curves, the pressurized thermal shock (PTS) and the low temperature overpressure mitigation system enable temperature settings.

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The fluence reevaluation accounted for the power uprate. The results indicated very small change in the projected fluence to the end of the current license [Reference 38]. Consequently, the range of validity of the current PT curves of 20 effective full-power years (EFPYs) remains unchanged.

Likewise, the RT_{PTS} value to the end of the current license remains well within the 10 CFR 50.61 limit of 270° F for plates and axial welds and 300° F for circumferential welds.

Because the projected value of the vessel fluence for 20 and 32 EFPYs remains unchanged, the above conclusions are acceptable.

3.2.12 Overpressure Protection

In a letter dated February 25, 2005 [Reference 11], the licensee provided the results of analyses to demonstrate that the SS safety valve design capacity continues to be sufficient to limit the pressure to less than 110% of the RCS pressure boundary design pressure (as specified by the ASME Code, Section III, Article NM-7000), during the most severe abnormal operational transient and the reactor scrammed, with sufficient available margin to account for uncertainties in design and operation of the plant.

The licensee performed the analyses incorporating assumptions consistent with those specified in NUREG-0800, SRP 5.2.2, Section II.A, including the assumption that reactor scram is initiated by the second safety-grade signal from the RPS. The NRC staff considered that the licensee used an upgraded computer model (RETRAN) from what had been used in the previous evaluations for these valves. This model is an acceptable model for these analyses. The difference in results assuming the second trip (overtemperature delta-T) versus the first trip (high pressurizer pressure) were small (<10 psia), and well below the 110% criterion, leaving sufficient available margin to account for uncertainties in design and operation of the plant. The staff finds the analyses acceptable because they were performed using an acceptable analysis model and analysis assumptions consistent with the guidance provided in SRP 5.2.2, Section II.A. The staff also finds that these analyses acceptably demonstrate that the SS safety valves continue to have sufficient capacity to satisfy their requirements, as stated above.

The pressurizer power operated relief valves were originally sized and evaluated for SS operation at the proposed uprated power conditions. Therefore, the proposed SPU will not change the existing evaluation or its acceptability. The staff find this acceptable.

The licensee's SPU submittal also evaluated the impact on the performance of the atmospheric dump valves. The licensee's analyses demonstrated that, for a 50% load rejection event, activation of the atmospheric dump valves would avert actuation of the associated steam line safety valves, the pressurizer relief valve, the pressurizer safety valves, and reactor trip.

The SS design to address low-temperature overpressure scenarios is addressed in the following section of this SE. The NRC staff review discussed in that section finds the SS low-temperature provisions acceptable.

In conclusion, the NRC staff finds that the overpressure protection provisions for SS at the uprated power acceptably address their functional requirements.

3.2.12.1 Cold Overpressure Mitigation System

The power uprate does not affect significantly the analysis or the results of cold overpressure mitigation. This analysis is performed for temperatures at or below 350° F. The power increase has none or a negligible effect on the results due to a small increase in decay heat and a 1% reduction in the reactor coolant volume due to an assumed 10% tube plugging. The PT limit curves remain the same. SS is equipped with two power-operated relief valves (PORVs) with adequate relief capacity to keep the cold vessel pressurization stresses below the 10 CFR Appendix G values. The PORV low temperature overpressure protection (LTOP) system enable settings are determined from the existing mass and heat injection transients, which are not affected by either vessel fluence or reactor core properties. The cold overpressure mitigation system settings remain unchanged and it is acceptable, because neither the vessel material properties nor the reactor parameters are affected by the power uprate.

3.2.13 Transient and Accident Analyses

The licensee re-analyzed the UFSAR Chapter 15 LOCA and non-LOCA transients and accidents in support of the SS SPU. These analyses were performed at a rated core power of 3587 MWt using plant parameter values for those operating conditions. The initial condition uncertainties were recalculated at power uprate conditions for use in the SS SPU program. These uncertainty calculations were performed for the uprate operating conditions based on the plant-specific instrumentation and plant calibration and calorimetric procedures. The staff reviewed the licensee's transient and accident analyses at the 3587 MWt SPU conditions to verify the acceptance criteria are still met under these conditions. The staff's review of the LOCA and non-LOCA transients and accidents is discussed in the following sections.

3.2.13.1 LOCA Analyses

The licensee described the SS LBLOCA and small-break LOCA (SBLOCA) analyses performed at the uprated power for 17x17 W robust fuel (with ZIRLO cladding) assemblies [References 1, 2, 3, 4, and 5]. The LBLOCA analyses were performed with the W LBLOCA methodology described in WCAP-12945-P-A [Reference 22]. The SBLOCA analyses' results were recalculated using the W SBLOCA methodology described in the W NOTRUMP (COSI) SBLOCA methodology. The NRC staff reviewed these analyses to assure that the licensee met the requirements of 10 CFR 50.46.

The licensee provided the LOCA plant-specific analyses results for the W fuel. The following table provides the licensee's LBLOCA analysis results.

*TABLE 1 - LOCA ANALYSIS RESULTS

Limiting Break Type/Size/location	LBLOCA/ Pump Discharge	SBLOCA/ 4-Inch Pump Discharge
Fuel Type	W 17 x 17 ZIRLO fuel	W 17 x 17 ZIRLO fuel
Peak Cladding Temperature (PCT)	1784 EF	1373 EF
Maximum Local Oxidation	3.53% *	0.20% *
Maximum Total Core-wide Hydrogen Generation (All Fuel)	(0.3%)*	(<<1.0%)*

*These LOCA local oxidation and core-wide hydrogen generator values [Reference 5] are bounding LOCA values for the fuel. The licensee states that operational controls are such that the total oxidation (including LOCA and pre-LOCA) will always be below 16%. The values for core-wide hydrogen generation do not include a pre-LOCA amount. This is reasonable because normal operational monitoring and procedures maintain operational (pre-LOCA) core-wide hydrogen at a very low level.

The calculated values given in the table above are less than the limits specified in 10 CFR 50.46(b) (1)-(3), which requires the PCT to be less than 2200 EF, the maximum cladding oxidation to be less than 17%, and the maximum hydrogen generation to be less than 1.0%. As a result, the licensee has demonstrated compliance with 10 CFR 50.46(b)(1)-(3). Additionally, the licensee, as discussed below, has demonstrated compliance with 10 CFR 50.46(b)(5). Inasmuch as no other consideration affects the SS core geometry, this assures that the SS core will remain amenable to cooling as required by 10 CFR 50.46(b)(4).

In summary, the NRC staff concludes that the licensee's LOCA analyses were performed with LOCA methodologies that apply to SS and demonstrate that it complies with the requirements of 10 CFR 50.46 (b)(1)-(5). Therefore, the NRC staff finds the licensee's LOCA analyses acceptable.

3.2.13.1.1 Overall Applicability of LOCA Analysis Methodologies

The W LBLOCA methodology [Reference 22] specifically applies to SS since it applies to all W 2-, 3- and 4-loop plant designs. SS is a 4-loop W design.

The licensee used the W NOTRUMP (with COSI) SBLOCA methodology to perform SBLOCA analyses for the SS power uprate. This methodology applies to all W 2-, 3- and 4-loop plant designs and, therefore, it is applicable to SS.

The licensee stated that both SS and its vendor (W) have ongoing processes which assure that the values and ranges of the LOCA analyses inputs for PCT-sensitive parameters conservatively bound the values and ranges of the as-operated plant for those parameters [Reference 5].

These LOCA methodologies apply to plants of W design and W fuels, and have no technical limitations which would preclude their use for the proposed SS power uprate. Further, the licensee's statement above provides the assurance that the analyses results obtained using those LOCA methodologies will continue to apply to SS. The staff concludes that W LOCA methodologies identified above apply to SS, which is a W-designed plant that uses W fuel.

3.2.13.1.2 Slot Breaks at the Top and Side of the Pipe

The staff requested that the licensee address slot breaks at the top and side of a reactor pump discharge cold leg pipe which could, under some circumstances, lead to greatly extended periods of core uncovery, resulting in fuel cladding oxidation in excess of the 10 CFR 50.46(b)(2) limit, and also possibly in excess of the total hydrogen limit of 10 CFR 50.46(b)(3). In its response, the licensee discussed information which is included in a generic W report written to address this issue [Reference 5]. In its response, the licensee also stated that the Emergency Operating Procedures (EOPs) at SS were based on approved Westinghouse Owners Group EOP guidelines, and they directed timely operator actions that would avoid the conditions for extended core uncovery. The licensee indicated that the operator procedures and actions would be effective in LOCA scenarios because extended core uncovery would take a significant amount of time to develop [Reference 5]. The licensee has concluded that the existing provisions continue to apply to the upcoming cycle of operation, because the extended core uncovery issue of concern is fuel-independent.

Based on its review of the information provided by the licensee, and as set forth above, the NRC staff concludes that the licensee's analysis has successfully addressed this issue. The resolution of this issue applies to the current SS licensing basis and does not resolve the generic issue of slot breaks at the top and side of the pipe for any vendor methodology.

3.2.13.1.3 Downcomer Boiling

The licensee provided the results of an analysis it had performed using the approved W best estimate LBLOCA methodology that demonstrate that following a LBLOCA SS would attain a stable and sustained core quench [Reference 5]. This indicates that, at SS, downcomer boiling would not occur to the extent that it would significantly degrade core cooling in the first 2000 seconds of an LBLOCA transient. Therefore, the NRC staff finds this acceptable. The NRC staff is presently pursuing concerns related to downcomer boiling in a generic matter. If that review raises any concerns applicable to the LOCA analyses at SS, then the NRC staff will request the licensee to address these issues consistent with any generic resolution.

3.2.13.1.4 Post-LOCA Long-Term Cooling (LTC)

The regulatory requirement for LTC is provided in 10 CFR 50.46(b)(5) which states "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." Although the SRP [Reference 7] provides some guidance, it essentially repeats the regulatory requirement. In practice, following successful calculated blowdown, refill, and reflood after initiation of larger LOCAs, and following the transient conditions that occur after initiation of smaller LOCAs, the LTC requirement will be met if the fuel cladding remains in contact with water so that the fuel cladding temperature remains essentially at or below the saturation temperature. A potential challenge to LTC is that boric acid (H_3BO_3) could accumulate within the RV, precipitate, and block water needed to keep the fuel cladding wetted by water. Consequently, the staff audited the licensee's approach to control H_3BO_3 during LTC.

The concern arises if a LOCA results in loss-of-water circulation through the core, such as may occur with a large cold leg break where ECCS water maintains water level above the bottom of the cold leg and no water leaves the RV via the hot legs due to the elevation of the flow path through the SG tubes. This results in boiling in the core, which provides core cooling, with steam leaving via the hot legs and passing through the SGs and out the break in the cold leg. Core water inventory removed by boiling is replenished via the downcomer due to the maintained cold leg water level. However, the incoming water contains H_3BO_3 and, since H_3BO_3 is not considered to be removed by the steam, H_3BO_3 will continue to concentrate in the core. Eventually, H_3BO_3 may begin to precipitate and could potentially block the flow of water needed to cool the fuel rods, raising the question of whether this meets the requirements of 10 CFR 50.46(b)(5). This condition is prevented by initiating injection of water into the hot legs at a rate greater than the boiloff rate so that water is forced into the lower plenum, up the downcomer, and out the cold leg break, thus preventing further increase in the H_3BO_3 concentration and preventing H_3BO_3 precipitation.

The licensee reported [Reference 1] that it analyzed this H_3BO_3 process using a variation of a model that was described in [Reference 24]. Analysis of H_3BO_3 behavior using a variation of the model of [Reference 24] was used for the Byron and Braidwood thermal power increase that was addressed by the staff via [Reference 25], where the staff found that certain assumptions [Reference 24] required modification. A further model variation to introduce conservatisms in the Byron and Braidwood analyses was described in [Reference 26] with staff approval via [Reference 27]. Although the licensee introduced some of the necessary modifications in its model, it is not clear that it addressed all necessary modifications. Thus, the licensee's statement regarding methodology [Reference 24], that "the methodology ... is consistent with, or otherwise conservative with respect to, the methodology described in" is not sufficient to justify the licensee's analysis.

Further, the models are limited to describing behavior associated with a large break LOCA, they do not fully represent H_3BO_3 behavior during reflood following initiation of the LOCA, they do not include consideration of potentially significant phenomena associated with transient or pseudo-steady state conditions, and they do not address potential behavior during smaller break-size LOCAs where natural circulation may be lost and regained, including whether H_3BO_3 may precipitate when cooler water circulates into the core following an extended time when H_3BO_3 may have been concentrating. Use of a model where such modeling considerations are not addressed is not unique to this licensee, and the staff has previously questioned H_3BO_3 behavior modeling during long term cooling when reviewing applications from other licensees.[1] In these cases, the NRC staff has considered the low probability that conditions leading to significant H_3BO_3 accumulation will be encountered and that there are a number of modeling conservatisms that tend to compensate for modeling inadequacies. Consequently, the NRC staff does not consider the outstanding issues to be a significant safety concern, and in the interim until generic concerns associated with LTC are resolved, for purposes of this review, the staff will rely on an interim evaluation of comparing LTC characteristics with cases where effective H_3BO_3 dilution action was initiated well before the staff judged the action was necessary.

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[1]In many analyses, H_3BO_3 has been assumed to concentrate in the volume contained in the core and in the volume of the upper plenum below the level of the bottom of the hot legs, with the assumption that that volume was filled with water and contained no voids. In a recent case, the staff and a licensee evaluated RCS response in which a significant void was calculated in the core and upper plenum so that the effective volume for dilution of H_3BO_3 was reduced from the typical assumption of a collapsed liquid level at the bottom of the hot leg. With this modeling, analyses established that steam flow would transport liquid out of the vessel via the hot legs through the SGs in the first hours following LOCA initiation. Further, in a realistic analysis, the volume in which H_3BO_3 was concentrating would also include the two-phase mixture in the hot legs, part of the upper plenum adjacent to the hot legs, and a portion of the lower plenum. These effects would reduce the H_3BO_3 concentration rate compared to the rate predicted if the two-phase mixture was assumed limited to the core and upper plenum below the hot legs. The timing predictions using this approach were found to be similar to those predicted by the historic models. Since conservatisms remain that are not credited in either of the approaches, the staff remains confident that more complete confirmatory analyses will establish that the approach to H_3BO_3 concentration will be shown to remain conservative.

This comparison is illustrated in the following table for Westinghouse-designed NSSS which compares typical plant characteristics:

Comparison of H_3BO_3 Accumulation Characteristics					
	Characteristic	Byron/ Braidwood 5% uprate	Kewaunee 6% uprate (7.4% including previous uprate)	Indian Point 2 3.26% uprate	Seabrook Station 5.2% uprate
1	Predicted time to reach H_3BO_3 saturation (hours)	8.53 (5/4/01) 6.0 (4/12/02)	7.8		
2	Power (MWt)	3587	1772 + 0.6% uncertainty	3216	3587 + 2% = 3659
3	Decay heat generation rate multiplier (dimensionless)	1 (5/4/01) 1.2 (4/12/02)	1	1.2	1.2
4	Assumed H_3BO_3 saturation limit (wt%)	23.53	23.53	23.53	23.53
5	Core plus upper plenum volume below hot leg (ft^3)	1072*	Power to volume ratio is similar between 2 and 4 loop Westinghouse plants	Same assumptions as used for Byron / Braidwood	Not specified
6	Minimum calculated time to hot leg injection based on flow rate needed for decay heat removal (hours)	-	-	-	4.0
7	Time to hot leg injection via emergency operating procedures (hours)	Consistent with Item 1 prediction	6.6	6.5	5.0 minimum 6.0 maximum
	References	24 - 27	39	40, 41, 42	1, 43, 28
*Value is from NUREG-1269, "Loss of Residual Heat Removal System, Diablo Canyon Nuclear Power Plant, Unit 2, April 10, 1987," June 1987.					

The staff notes that the Byron/Braidwood licensees calculated 6 hours (April 12, 2002, see Table) when the allowable H_3BO_3 concentration would be reached, whereas SS calculated 7.46 hours. The licensee's prediction also appears to be inconsistent with the predictions for Kewaunee and Indian Point. For example, if SS and Indian Point were identical except for thermal power, the Indian Point 6.76 hours would be reduced to [3659/3216] = 5.94 hours, a value consistent with the Byron/Braidwood prediction of 6 hours, but one that is inconsistent with the SS prediction of 7.46 hours. The licensee did not provide sufficient information in its request to allow the NRC staff to determine the source of the apparent inconsistency. However, the NRC staff notes that the licensee did not rely solely upon its prediction to determine operator action times in its EOPs. Rather, the licensee specified conservative times of 5 hours minimum and 6 hours maximum for initiation of hot leg injection [Reference 28]. The 6-hour specification is consistent with values previously accepted by the NRC staff as adequate for ensuring establishment of effective hot leg injection in similar plants and the 5-hour specification is conservative with respect to the licensee's 4 hour prediction of the minimum time that can elapse before initiation of hot leg injection. Further, the licensee estimated that it would take approximately 10 minutes to accomplish the switchover action. The 1-hour time span specified in the procedures is conservative when compared to the time needed to accomplish the action.

Therefore, while the NRC staff cannot endorse the licensee's evaluation as a valid mechanistic model of the phenomena, the staff believes, on an interim basis, that there is sufficient basis to approve the license amendment with respect to LTC and potential H_3BO_3 precipitation concerns.

This NRC staff conditional acceptance will remain effective until generic concerns associated with LTC are rectified, at which time the licensee will have to establish that it is in compliance with the resolution of the generic concerns.

3.2.13.2 Non-LOCA Transients and Accidents

The licensee re-analyzed SS's UFSAR Chapter 15 non-LOCA events at the SPU conditions of 3587 MWt. The licensee used the NRC previously-approved computer codes and methodologies for each of the non-LOCA transient analyses at SS. The licensee used the RETRAN computer code in the SS non-LOCA SPU safety analyses, simulating a W 4-loop plant design, applicable to SS, as described and presented in WCAP-14882-P-A [Reference 32]. The licensee used RETRAN in combination with VIPRE-01 for reactor core subchannel thermal-hydraulic calculations, a neutronic code such as ANC, and a fuel performance code such as FACTRAN in core design, as described in [References 20, 15, 16, and 30], respectively. The licensee used TWINKLE [Reference 29], a multidimensional neutron computer code, in conjunction with FACTRAN [Reference 30], a code for thermal transients in a UO_2 fuel rod, to perform the rod cluster control assembly (RCCA) ejection and uncontrolled RCCA withdrawal from a subcritical or low power startup condition analyses. The licensee met the conditions and restrictions set on the specific codes. Where applicable, the licensee used the previously-approved RTDP methodology discussed in WCAP-11397-P-A [Reference 17] in performing the non-LOCA safety analyses. The staff finds the codes and methodologies used by the licensee to perform the safety analyses under SPU conditions acceptable since the licensee satisfies the conditions and restrictions set on the specific codes for application at SS. Table 6.3.1-1 and Table 6.3.1-2 in the licensee's application [Reference 1] provide non-LOCA selected analyses results and non-LOCA plant initial conditions, respectively, for the SS SPU conditions [Reference 1].

3.2.13.2.1 Excessive Heat Removal Due to FW System Malfunctions and Increase in FW Flow or Decrease in FW Temperature

A change in SG FW conditions that results in an increase in FW flow or a decrease in FW temperature could result in excessive heat removal from the RCS. Such changes in FW flow or FW temperature are a result of a failure of a FW control valve or FW bypass valve, failure in the FW control system, or operator error. Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The acceptance criteria are based on critical heat flux not being exceeded, pressure in the RCS and main steam system (MSS) being maintained below 110% of the design pressures, and the peak linear heat generation rate not exceeding a value that would cause fuel centerline melt. Specific review criteria are contained in SRP Section 15.1.1-4.

The licensee used the RETRAN computer code [Reference 32] to analyze the excessive heat removal due to a FW system malfunction. The VIPRE subchannel code [Reference 20] calculated the hot channel heat flux transient and departure from nucleate boiling ratio (DNBR). The RPS provided mitigation for this event and the results showed RCS pressure remained below the 110% design value. The limiting case DNBR value remained above the SAL of 1.47 listed in Table 6.3.1-1 [Reference 1].

The staff reviewed the licensee's analysis and concludes that the licensee's analysis was performed using acceptable analytical models. The staff finds the licensee demonstrated that the reactor protection and safety systems will continue to ensure the critical heat flux will not be exceeded, pressure in the RCS and MSS will be maintained below 110% of the design pressures, and the peak linear heat generation rate will not exceed a value that would cause fuel centerline melting. The staff concludes that SS will continue to meet GDC-10, 15, 20 and 26 following implementation of the proposed SPU. Therefore, the staff finds the proposed power uprate acceptable with respect to the excessive heat removal due to FW system malfunction event.

3.2.13.2.2 Excessive Increase In Main Steam Flow

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the SG load demand. The RCS is designed to accommodate a 10% step-load increase or a 5% per minute ramp load increase in the range of 15 to 100% of full power, taking credit for all controls systems in automatic. Any loading rate in excess of these values can cause a reactor trip actuated by the RPS. The acceptance criteria are based on critical heat flux not being exceeded, pressure in the RCS and MSS being maintained below 110% of the design pressures, and the peak linear heat generation rate not exceeding a value that would cause fuel centerline melt.

The licensee used the RETRAN computer code [Reference 32] to analyze this event. The licensee evaluated four cases which demonstrated that the fuel cladding integrity will not be adversely affected following a 10% step-load increase from rated load. Two cases were analyzed, assuming manual rod control, with BOL and end-of-life (EOL) reactivity feedback. An additional two cases were analyzed, assuming automatic rod control, also with BOL and EOL reactivity feedback. The RPS was assumed to operate, as designed, in all the case studies. In performing its evaluation, the licensee used conservatively bounding conditions in generating statepoints using the RTDP methodology, which are then compared directly to the SS SPU core limits. The licensee evaluated the effect of this transient on the minimum DNBR by applying conservative deviations on the initial conditions for core power, vessel average temperature, and pressurizer pressure at the normal full power operating conditions in order to generate a limiting set of statepoints. The bounding deviation values in plant parameters were used in the evaluation of this transient. These deviations bound the range of variations that could occur as a result of an excessive load increase accident and were applied only in the direction that would have the most adverse impact on DNBR (i.e., to minimize the DNBR calculated value). The statepoints generated were compared to the SS SPU limiting DNB core limit lines that represent the limiting DNBR conditions for the uprate. The licensee found that when applying conservatively bounding conditions to the plant parameters for this event, the corresponding minimum DNBR statepoint conditions remained above the SPU DNBR SAL.

The staff reviewed the licensee's evaluation of the excessive load increase incident and concludes that the licensee's analysis demonstrated the SPU DNBR SAL remains bounding for this event. The staff finds the licensee demonstrated that the reactor protection and safety systems will continue to ensure critical heat flux will not be exceeded, pressure in the RCS and MSS will be maintained below 110% of the design pressures, and the peak linear heat generation rate will not exceed a value that would cause fuel centerline melt. The staff concludes that SS will continue to meet GDC-10, 15, and 26 following implementation of the proposed SPU and finds the proposed power uprate acceptable with respect to the excessive load increase incident.

3.2.13.2.3 Inadvertent Opening of a SG Dump Relief or Safety Valve

The inadvertent opening of a SG dump, relief, or safety valve is an AOO (or ANS Condition II event). The acceptance criteria for ANS Condition II events do not allow for any fuel failure. Accident analysis results, which indicate that the DNB SAFDL is not violated, demonstrate that no fuel failures are predicted to occur, due to this event.

The licensee has not analyzed this event. Instead, the licensee has analyzed the zero power steamline rupture [Reference 1], an ANS Condition IV event that results in a greater release of steam from the SGs, and causes a more severe cooldown of the core. The results of this analysis indicate that the DNB SAFDL is not violated (i.e., ANS Condition II acceptance criteria are satisfied).

The licensee maintains that the inadvertent opening of a SG dump, relief, or safety valve would cause a slower SG blowdown and core cooldown than would the steamline rupture event. Any resulting return to power would not be as high as the return to power predicted for the zero-power steamline rupture. Accordingly, the minimum DNBR caused by the inadvertent opening of a SG dump, relief, or safety valve would be higher than the minimum DNBR resulting from the zero-power steamline rupture. The consequences of the inadvertent opening of a SG dump, relief, or safety valve are bounded by the consequences of the zero-power steamline rupture.

Since the analysis results of the zero-power steamline rupture indicate that the minimum DNBR remains above the SAL, then the minimum DNBR resulting from the inadvertent opening of a SG dump, relief, or safety valve would also remain above the SAL. No fuel failure would be predicted for either the zero-power steamline break or the inadvertent opening of a SG dump, relief, or safety valve.

Such a comparison between ANS Condition II and ANS Condition IV events is possible since both events meet the same, ANS Condition II acceptance criteria. The staff agrees with this reasoning and conclusion.

3.2.13.2.4 Steam System Piping Failure

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. Reactor protection and safety systems are actuated to mitigate the transient. The core is shut down by the H_3BO_3 injection into the RCS by the SIS. Since this is an ANS Condition IV event, fuel failure may occur provided that any resulting release of radioactive material does not exceed the guidelines of 10 CFR Part 100.

In performing its analysis, the licensee assumed the most reactive RCCA stuck in its fully withdrawn position. The licensee reviewed two cases, one with offsite power available, and the other with loss of offsite power (LOOP). The limiting steamline break with return to power corresponds to the case with offsite power available which results in the challenge to DNB. The licensee used the RETRAN computer code [Reference 32] to calculate the core heat flux and the RCS temperature and pressure resulting from the cooldown. The licensee considered the key analysis assumptions to maximize the cooldown of the RCS, so as to maximize the positive reactivity insertion, and thus maximize the peak return to power. The licensee performed the analysis using the VIPRE code [Reference 20] to determine if the DNBR fell below the SAL. The licensee performed the DNBR analysis for the most conservative case and found that the resulting DNBR was 1.79 which remained above the DNBR SL of 1.47 for this event under SPU conditions.

The staff reviewed the licensee's analysis of the excessive heat removal due to steamline break and concludes that the licensee's analysis was performed using acceptable analytical models. The staff concludes the licensee met the DNB design basis criterion and finds the proposed power uprate acceptable with respect to the steamline break.

3.2.13.2.5 Loss of External Load/Turbine Trip

A major loss-of-load can result from either a loss-of-external electrical load or from a turbine trip from full power without a direct reactor trip. The turbine-trip event is more severe than the total loss-of-external-electrical-load event since it results in a more rapid reduction in steam flow. These events result in a sudden reduction in steam flow and, consequently, pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The acceptance criteria are based on the minimum DNBR remaining above the SAL and pressure in the RCS and MSS remaining below 110% of the design pressures values. Specific review criteria are contained in SRP Section 15.2.1-5.

The licensee used the RETRAN computer code [Reference 32] to analyze this event. The licensee re-analyzed the minimum DNBR case with automatic pressurizer pressure control taking credit for the effect of the pressurizer spray and power-operated relief valves in reducing or limiting the primary coolant pressure. The RCS peak pressure case did not consider automatic pressurizer pressure control. No credit was taken for the effect of the pressurizer spray or power-operated relief valves in reducing or limiting the primary pressure. For MSS overpressure concern, automatic pressurizer pressure control was assumed available. In performing its analyses, the licensee assumed minimum reactivity feedback (BOL) conditions and a least negative Doppler power coefficient and no credit for operation of the steam dump system or SG atmospheric valves, which maximizes secondary pressure. Additionally, the licensee assumed main FW flow was terminated at the time of the turbine trip, with no credit taken for emergency FW system, except for long-term recovery to mitigate the consequences of the transient. The licensee used the STDP methodology to analyze the peak pressure cases and the RTDP methodology to analyze the minimum DNBR case. The licensee performed the analyses using the RETRAN computer code to determine the plant transient conditions following a total loss-of-load for both conditions. The reactor tripped on a high-pressurizer pressure trip signal. The pressurizer water-solid condition was precluded, thus not compromising the RCS pressure boundary and preventing progression into another condition event. The results showed the primary system pressure (2681.9 psia) which remained below the 110% design pressure value (2748.5 psia) and the secondary side steam pressure

(1302.8 psia) which remained below 110% of the SG shell design pressure (1318.5 psia). The minimum DNBR with pressure control case took credit for the pressurizer spray and pressurizer PORVs, but not the steam dump. The reactor tripped on a high pressurizer pressure reactor trip signal. The analysis results showed the minimum DNBR (1.83) which remained above its SAL of 1.47 under the SPU conditions. Therefore, no core SL will be violated as a result of implementing the SPU.

The staff reviewed the licensee's analyses of the loss of external electric load and concludes that the licensee's analyses were performed using acceptable analytical models, as stated above. The staff finds that the licensee demonstrated that the minimum DNBR will remain above the SAL and pressure in the RCS and MSS will remain below 110% of the design pressure values for the proposed power uprate. The staff concludes that SS will continue to meet GDC-10, 15, and 26 following implementation of the proposed SPU conditions. Therefore, the staff finds the proposed power uprate acceptable with respect to the loss of external electric load.

3.2.13.2.6 Loss of Normal Feedwater (LONF) Flow

A LONF flow could occur from pump failures, valve malfunctions, or a LOOP. LONF flow results in an increase in reactor coolant temperature and pressure, which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a LONF flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The acceptance criteria are based on the minimum DNBR remaining above the SAL, pressure in the RCS and MSS being maintained below 110% of the design pressures and the pressurizer is prevented from becoming water-solid. Specific review criteria are contained in SRP Section 15.2.7.

The licensee used the RETRAN computer code [Reference 32] to analyze this event. The analysis was performed to show that following a LONF, the emergency FW system is capable of removing the stored energy, residual decay heat and reactor coolant pump heat. The loss of FW event is bounded by the loss of load/turbine trip event analysis for overpressurization concerns and DNBR since the turbine trip event is the initiating event and the loss of heat sink is much more severe in comparison to loss of FW event. In performing its analysis, the licensee used conservative assumptions to maximize the time to reactor trip and to minimize the energy removal capability of the emergency FW system. The licensee assumed the RCPs operated continuously throughout the transient, the initial RCS average temperature assumed to be 584.1 EF, and the initial pressurizer water level assumed to be 65% level. The pressurizer spray, PORVs, and heaters were assumed to be operable to maximize the pressurizer water volume. The reactor trip occurred after 77 seconds on SG low-low water level, and emergency FW system flow of 650 gpm is initiated from one emergency FW pump, with flow split equally between four SGs. The worst single failure modeled in the analysis is the loss of one emergency FW pump. The results of the analysis showed that the pressurizer did not reach a water-solid condition. The calculated long-term peak pressurizer water volume was 1175 cubic feet compared to the total pressurizer volume limit of 1834.4 cubic feet. The analysis performed also showed the peak RCS and MSS pressures remained below the 110% design pressures throughout the transient. With respect to DNB, the LONF accident was bounded by the loss of load accident. For the LONF transient, the RCS temperature increases gradually as the SGs boil down to the low-low level TSP, at which time the reactor trips and immediately after, the turbine trips. Nuclear power drops at nearly the same time steam flow drops and there is very little mismatch between the primary and secondary systems to force an RCS heatup. For the loss-of-load transient, the turbine trip is the initiating event, and the power mismatch between the primary and secondary systems is more severe. The RCS heatup will be much more severe for the loss-of-load transient than the LONF transient, in which case the loss-of-load transient demonstrated the minimum DNBR remained greater than the SAL.

The staff reviewed the licensee's analysis for the LONF flow transient and concludes that the licensee's analysis has adequately accounted for operation of the plant at the proposed power level and was performed using an acceptable analytical model. The staff finds the licensee demonstrated that the minimum DNBR SAL will not be exceeded, pressure in the RCS and MSS will be maintained below 110% of the design pressures, and a more serious plant condition is precluded. The pressurizer would not become water-solid during this transient and the one emergency FW system capacity is sufficient to dissipate core residual heat, stored energy, and reactor coolant pump heat such that reactor coolant would not discharge water through the pressurizer relief or safety valves. The staff concludes that SS will continue to meet GDC-10, 15, and 26 following implementation of the proposed SPU. Therefore, the staff finds the proposed SPU acceptable with respect to the LONF flow event.

3.2.13.2.7 LOOP

The loss of non-emergency ac power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant pumps. This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops (RCLs) which was supported by the emergency FW system in the secondary system. Reactor protection and safety systems are actuated to mitigate the transient. The acceptance criteria are based on the minimum DNBR remaining above the SAL, pressure in the RCS and MSS being maintained below 110% of the design pressures. Specific review criteria are contained in SRP Section 15.2.6.

This transient was analyzed using the RETRAN computer code [Reference 32]. From its analysis, the licensee concluded that in a loss of ac (LOAC) power to the station auxiliaries, the plant response is almost identical to the complete loss of reactor coolant flow event at SS. After the reactor trip, the emergency FW system removes decay heat and this portion of the transient is similar to the LONF event. The RETRAN code results showed that natural circulation and the emergency FW flow available were sufficient to provide adequate core decay heat removal following a reactor trip and RCP coastdown. The results also showed the calculated long-term peak water volume in the pressurizer was 1463 cubic feet compared to a total volume limit of 1834.4 cubic feet. Therefore, the pressurizer did not reach a water-solid condition and the pressurizer relief and safety valves would not relieve reactor coolant. The RCS and MSS pressures remained below the applicable design limits throughout the transient. The licensee stated the LOAC power event was bounded by the complete loss of reactor coolant flow event since the first few seconds of the transient would be almost identical to the complete loss of reactor coolant flow event, in which pump coastdown inertia along with the reactor trip prevents reaching the DNBR SAL. Since this is a heatup transient, DNB is not challenged during this event.

The staff reviewed the licensee's analysis of the LOAC power to plant auxiliaries and concludes that the licensee's analysis was performed using an acceptable analytical model, as stated above. The staff finds the licensee demonstrated that the reactor protection and safety systems will continue to ensure that the specified fuel design limits are not exceeded, the peak primary and secondary system pressures are not exceeded, and a more serious plant condition is precluded. The staff concludes that the plant will continue to meet GDC-10, 15 and 26 following implementation of the proposed power uprate. Therefore, the staff finds the proposed power uprate acceptable with respect to the LOAC power to the plant auxiliaries.

3.2.13.2.8 FW System Pipe Breaks

A major FW line rupture is defined as a break in a FW line large enough to prevent the addition of sufficient FW to the SGs to maintain shell-side fluid inventory. Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive discharge of steam through the break) or a RCS heatup. Cases that can cause a RCS cooldown are covered by the analysis of the steamline break event. Therefore, FW line rupture is evaluated as one of the events that can cause a RCS heatup.

Analysis of this event demonstrates the ability of the emergency feedwater system to remove core decay heat and thereby ensure that the core remains in a coolable geometry. It is inferred that the core remains covered with water (and coolable) by showing that the hot and cold leg temperatures remain subcooled until the emergency FW heat removal rate exceeds the RCS heat generation rate (mainly from decay heat). The analysis also demonstrates that the primary and secondary system pressures remain within 110% of their design pressures.

The FW line rupture event is classified as an ANS Condition IV event, a limiting fault, as defined by the ANS. ANS Condition IV events are limiting faults that are not expected to occur, but are postulated because their consequences would include the potential for release of significant amounts of radioactive material. Guidelines for the staff's review of the FW line rupture event are provided in Section 15.2.8 of the SRP [Reference 7].

The RETRAN computer code [Reference 32] was used to calculate the power transient and the associated temperatures of the reactor coolant various locations in the RCS. These are compared to the saturation temperature, which is based upon the RCS pressure. Demonstration that the hot and cold leg temperatures remain below the saturation temperature, implies that the core remains covered throughout the transient. The major assumptions of the analysis are selected to conservatively maximize the RCS fluid temperatures and minimize the saturation temperature.

Four FW line rupture analysis cases were considered: two maximum reactivity feedback cases, with and without offsite power available and two minimum reactivity feedback cases, with and without offsite power available. The maximum reactivity feedback cases assumed most positive moderator density coefficients, most negative Doppler temperature coefficients, most negative Doppler-only power coefficients and minimum delayed neutron beta-effective values. The minimum reactivity feedback cases assumed zero moderator density coefficients, least negative Doppler temperature coefficients, least negative Doppler-only power coefficients and maximum delayed neutron beta-effective values. The minimum reactivity feedback case with offsite power available was seen to be the most limiting case.

The primary and secondary systems were calculated to remain below 110% of their respective design pressures. After the reactor trip, the RCS heats up and pressurizes until the pressurizer power operated relief valves open and the heat removal rate, due to steam relief through the main steam safety valves (MSSVs) and emergency FW injection, exceeds the core decay heat plus reactor coolant pump heat. When this point is reached, temperatures begin to decrease and the adequacy of the Emergency Feedwater System is demonstrated. Since the maximum hot and cold leg temperatures remain below the saturation temperature throughout the transient, it is demonstrated that the core remains covered and coolable.

The staff agrees that the postulated FW line rupture analysis indicates that the Emergency Feedwater System capacity is adequate to remove decay and reactor coolant pump heat, and to prevent uncovery of the reactor core, under SPU conditions.

3.2.13.2.9 Partial Loss of Forced Reactor Coolant Flow

The partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump or from a fault in the power supply to the reactor coolant pump. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB, with subsequent fuel damage.

A partial loss of coolant flow may be caused by a mechanical or electrical failure in a pump motor, a fault in the power supply to the pump motor, or a pump motor trip caused by such anomalies as over-current or phase imbalance. SS's four reactor coolant pumps are supplied through two busses. The licensee's partial loss of coolant flow accident analysis postulates a failure that causes two reactor coolant pumps to coast down.

The partial loss-of-flow event is categorized as an ANS Condition II event. It is analyzed to demonstrate that the DNBR remains above the SAL value, and that the peak RCS and secondary system pressures remain below their respective design limits.

The RETRAN computer code [Reference 32] is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE computer code [Reference 20] is then used to calculate the hot-channel heat flux transient and DNBR, based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN.

This event is analyzed following the RTDP [Reference 17], which assumes that initial reactor power, pressurizer pressure, and RCS temperature are at their nominal values. The MMF is also assumed. Assumptions are made such that the core power is maximized during the initial part of the transient when the minimum DNBR is reached.

The analysis results indicate that ANS Condition II acceptance criteria are satisfied, and particularly that the minimum DNBR remains above the SAL value. Thus, no fuel failures are predicted. The staff agrees with the licensee's approach and results.

3.2.13.2.10 Total Loss of Forced Reactor Coolant Flow

A complete loss of forced reactor coolant flow could result from a simultaneous loss of electrical supplies to all reactor coolant pumps. A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer and a subsequent increase in fuel temperature. Accompanying fuel damage could then result if SAFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The acceptance criteria are based on the minimum DNBR remaining above the SAL, and pressure in the RCS and MSS being maintained below 110% of the design pressures. Specific review criteria are contained in SRP Section 15.3.1-2.

The licensee re-analyzed the complete loss of reactor coolant flow at SPU conditions. The licensee used the RTDP methodology [Reference 17], the RETRAN code [Reference 32] and VIPRE code [Reference 20]. For the partial loss of flow incident, the DNBR did not decrease below the SAL at any time during the transient (see Section 3.2.13.2.9). The peak primary and secondary system pressures remained below their respective limits at all times. For the complete loss-of-flow event, the licensee analyzed both undervoltage and underfrequency decay transients. The VIPRE analyses for these scenarios confirmed that the minimum DNBR values (1.998) were greater than the SAL of 1.47. The peak primary and secondary system pressures remained below their respective limits at all times. The results of the licensee's analyses demonstrated that the acceptance criteria for these events were satisfied.

The staff reviewed the licensee's analyses of the complete loss of reactor coolant flow and concludes that the licensee's analyses were performed using acceptable analytical models. The staff finds the licensee demonstrated that the reactor protection and safety systems will continue to ensure the minimum DNBR will remain above the SAL and pressure in the RCS and MSS will be maintained below 110% of the design pressures. The staff further concludes that the licensee will continue to meet GDC-10, 15, and 26 following implementation of the proposed SPU. Therefore, the staff finds the proposed power uprate acceptable with respect to the complete loss of reactor coolant flow.

3.2.13.2.11 Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

In a locked rotor accident, the events postulated are an instantaneous seizure of the reactor coolant pump rotor or the break of the shaft of a reactor coolant pump in a pressurized-water reactor (PWR). Flow through the affected RCL is rapidly reduced, leading to a reactor trip on a low flow signal. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The locked rotor accident is an ANS Condition IV event and the acceptance criterion is based on rods-in-DNB being less than or equal to that assumed in the radiological dose analyses for the locked rotor/shaft break event.

The licensee re-analyzed the locked rotor accident using the most limiting combination of conditions for the locked rotor and pump shaft break events with a total of four loops in operation. The first case used the STDP methodology to evaluate the RCS pressure and fuel clad temperature transient conditions. The second case used the RTDP methodology [Reference 17] to evaluate DNB in the core during the transient. The licensee performed the analyses using the RETRAN computer code [Reference 32] to calculate the loop and core flow transients, nuclear power transient, and primary system pressure and temperature transients. The VIPRE subchannel code [Reference 20] calculated the hot channel heat flux transient and DNBR, based on the nuclear power and RCS flow from RETRAN. The DNBR value remained above the SAL. The peak RCS pressure was 2544 psia which is less than the acceptance criterion of 2750 psia. The PCT was 1650 EF which was considerably less than the limit of 2700 EF for this event. For radiological dose evaluation, no fuel rods experienced DNB (rods-in DNB case).

The staff reviewed the licensee's analyses of the locked rotor and pump shaft break events and concludes that the licensee's analyses were performed using acceptable analytical models. The staff concludes that the plant will continue to meet GDC-27, 28, and 31 following implementation of the proposed power uprate. Therefore, the staff finds the proposed power uprate acceptable with respect to the locked rotor accident.

3.2.13.2.12 Uncontrolled RCCA Withdrawal from Subcritical

The RCCA withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA banks resulting in a power excursion.

This event is defined to occur while the core is in a subcritical condition. Therefore, it is not expected that this event would be materially affected by an increase in rated power level. The licensee has presented an analysis, in the license amendment request (LAR), which indicates that all the ANS Condition II acceptance criteria are met.

The staff agrees with the results of this analysis.

3.2.13.2.13 Uncontrolled RCCA Withdrawal at Power

The uncontrolled RCCA bank withdrawal at power event is an ANS Condition II event that is defined as the inadvertent addition of reactivity to the core caused by the withdrawal of RCCA banks when the core power level is zero or greater (Mode 1 - power operation). The reactivity insertion resulting from the bank (or banks) withdrawal will cause an increase in core nuclear power and subsequent increase in core heat flux.

The RETRAN code [Reference 32] was used to analyze this event. A series of cases is evaluated: each of which consists of inserting reactivity, at a constant rate, until an automatic reactor trip occurs. The cases address a spectrum of possible reactivity insertion rates up to a maximum positive reactivity insertion rate that is greater than would be reasonably experienced during normal operation. The power mismatch and resultant temperature rise could eventually result in DNB and/or fuel centerline melt. Additionally, the increase in RCS pressure, associated with the increase in temperature, could challenge the integrity of the RCS pressure boundary or the MSS pressure boundary.

The RPS is designed to automatically terminate the event before the DNBR falls below the SAL value, the fuel rod KW/ft limit is exceeded, the peak pressures exceed their respective limits, or the pressurizer fills.

The principal parameter of interest is the DNBR, and principal means of protection lie in the power-range high neutron flux and the overtemperature ΔT trip logic. Other trips that may occur, during this event are the high nuclear flux trip, the overpower ΔT trip, the high pressurizer pressure trip, and the high pressurizer water level trip.

Raising the rated power level to SPU conditions may require adjusting some of the RPS TSPs. The uncontrolled RCCA bank at power event is one of the events that are used to determine setpoints for the RPS, particularly the overpower and overtemperature ΔT trips.

The staff concurs with the licensee’s analyses of the uncontrolled RCCA bank withdrawal at power event and agrees that the analysis results and conclusions are compatible with the proposed SPU.

3.2.13.2.14 RCCA Misoperation

The NRC staff's review covered the types of control rod misoperations that are assumed to occur, including those caused by a system malfunction or operator error. The review covered (1) descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations; (2) the sequence of events; (3) the analytical model used for analyses; (4) important inputs to the calculations; and (5) the results of the analyses. The NRC’s acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to initiate automatically operation of systems and components important to safety under accident conditions; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.3 and further guidance is provided in Matrix 8 of RS-001.

3.2.13.2.14.1 Technical Evaluation

The licensee addressed several RCCA misoperation events, including:

- Dropped full-length RCCAs
- Dropped full-length RCCA banks
- Statically misaligned full-length RCCAs
- Withdrawal of a single RCCA
- Dropped full length RCCAs: The transient response analysis, nuclear peaking factor analysis, and performance of the DNB design basis confirmation are performed in accordance with an approved methodology [Reference 9].

A generic statepoint analysis for this event [Reference 10], which was performed to bound a number of four-loop PWRs, was evaluated and determined to remain applicable for the SPU. With the generic statepoints being applicable, the effects of the SPU are accounted for when performing the nuclear and DNB analyses, which are performed on a cycle-specific basis.

- Dropped full-length RCCA banks: A dropped RCCA bank results in a symmetric power change in the core. Assumptions made in the methodology [References 9 and 10] for the dropped RCCAs analysis provide a bounding analysis for the dropped RCCA bank.

- Statically misaligned full length RCCAs: Steady-state power distributions are analyzed using the appropriate nuclear physics computer codes. The peaking factors are then compared to peaking factor limits developed using the VIPRE code, which are based on meeting the DNBR design criterion. The following cases are examined in the analysis assuming the reactor is at full power: the worst rod withdrawn with bank D inserted at the insertion limit, the worst rod dropped with bank D inserted at the insertion limit, and the worst rod dropped with all other rods out. It is assumed that the incident occurs at the time in the cycle with maximum predicted peaking factors. This assures a conservative F.ΔH for the misaligned RCCA configuration.

- Withdrawal of a single RCCA: Power distributions within the core are calculated. The peaking factors are then used by VIPRE to calculate the DNBR for the event. The case of the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power, was analyzed. This incident is assumed to occur at BOL since this results in the minimum value of the MTC. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

3.2.13.2.14.2 Summary

The NRC staff has reviewed the licensee's analyses of control rod misoperation events and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models.

The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs will not be exceeded during normal or anticipated operational transients. Based on this, the NRC staff concludes that SS will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to control rod misoperation events.

3.2.13.2.15 Startup of an Inactive Reactor Coolant Pump

Since operation with an idle reactor coolant pump is precluded by SS's TSs, the licensee maintains that this event need not be considered. The staff agrees.

3.2.13.2.16 Inadvertent Boron Dilution

Unborated water can be added to the RCS, via the chemical and volume control system (CVCS). This may happen inadvertently because of operator error or CVCS malfunction and cause an increase in reactivity and a decrease in shutdown margin. The CVCS system is designed to limit the potential rate of dilution to a value that, after indication through alarms and instrumentation, provides the operator with sufficient time to correct the situation in a safe manner. The operator must stop this unplanned dilution before the shutdown margin is lost. The acceptance criteria are based on the core integrity and overfilling of the pressurizer. Specific review criteria are contained in SRP Section 15.4.6.

The licensee considered three different categories of CVCS malfunction transients: (1) malfunctions that result in the injection of water with a boron concentration greater than the RCS boron concentration, (2) malfunctions that result in the injection of water with a boron concentration less than the RCS boron concentration, and (3) malfunctions that result in the injection of water with a boron concentration equal to the RCS boron concentration.

CVCS malfunctions of the first category are bounded by the inadvertent safety injection (SI) actuation analysis (see Section 3.2.13.2.19). Events of the second category, also known as boron dilution, are bounded by the analyses of the UFSAR [Reference 8].

The licensee re-analyzed the CVCS malfunction transient which results in the injection of water with a boron concentration equal to the RCS boron concentration. The staff's evaluation of this case analysis is provided in Section 3.2.12.2.20.

3.2.13.2.17 Inadvertent Loading of a Fuel Assembly into an Improper Position

This event, classified as an ANS Condition III incident, encompasses the inadvertent loading of one or more fuel assemblies into improper positions or the loading of a fuel rod with one or more pellets of the wrong enrichment. Such errors can cause unexpected distortions in the core power shapes. Reference [1] indicates there is a 5% uncertainty margin included in the design value of the power peaking factor assumed in the analysis of ANS Condition I and ANS Condition II transients.

Reload startup physics tests can detect abnormal power shapes, which could be used to prevent operation of the plant with an error in core loading. Also, the fabrication of fuel rods and assemblies, and the core loading sequence are controlled by strict administrative procedures.

The staff agrees that any power distribution anomalies will either be detected by the reload startup test program or will be small enough to fall within the uncertainties allowed between nominal and design power shapes, even under SPU conditions.

3.2.13.2.18 RCCA Ejection Accidents

Control rod ejection accidents cause a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage. The NRC staff evaluated the consequences of a control rod ejection accident to determine the potential damage caused to the RCPB and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. The NRC staff's review covered initial conditions, rod patterns and worths, scram worth as a function of time, reactivity coefficients, the analytical model used for analyses, core parameters which affect the peak reactor pressure or the probability of fuel rod failure, and the results of the transient analyses. The NRC's acceptance criteria are based on GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other RVIs so as to impair significantly the capability to cool the core. Specific review criteria are contained in SRP Section 15.4.8.

The RCCA ejection event, an ANS Condition IV transient, is evaluated by performing a neutron kinetic analysis, and feeding the results into a hot-spot fuel heat transfer analysis. The overpressure transient is addressed separately, on a generic basis.

The licensee used a 1-D axial kinetics model, TWINKLE [Reference 29] to calculate the core nuclear power transient, including the various total core feedback effects from Doppler reactivity and moderator reactivity. The average core nuclear power is multiplied by the post-ejection hot-channel factor, and the fuel enthalpy and temperature transients at the hot spot are calculated with the detailed fuel and cladding transient heat transfer computer code, FACTRAN [Reference 30].

This methodology, described in WCAP-7588, Rev. 1-A [Reference 31], has been applied to analyze the overpressurization transient of the RCS and determine the number of rods-in-DNB, as a result of a postulated ejected rod, on a generic basis for W PWRs. Overpressurization is addressed by calculating the pressure surge for an ejection worth of one dollar at BOL, hot full power. The results indicate that the peak pressure is less than the pressure that would cause stresses in the RCS to exceed their faulted condition stress limits. SS would be covered by this worst-case analysis. Therefore, the RCCA ejection accident would not overpressurize or damage the RCS.

A detailed three-dimensional calculation [Reference 31] also establishes an upper-limit to the number of rods-in-DNB, caused by the RCCA ejection accident, at 10%. SS is also encompassed by this worst-case analysis. The maximum number of rods-in-DNB following an RCCA ejection for SS would be less than 10%. This is less than 15% used in the radiological dose evaluation. Since the maximum number of fuel rods experiencing DNB is limited to 15%, the fission product release will not exceed the limits of 10 CFR 50.67.

The NRC staff has reviewed the licensee's analyses of the rod ejection accident and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage that would significantly impair the capability to cool the core. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC-28 following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to the rod ejection accident.

3.2.13.2.19 Inadvertent Actuation of the ECCS

Inadvertent actuation of the ECCS is an ANS Condition II event that is evaluated for the effects of adding water inventory to the RCS. This event could be caused by operator error or a spurious actuating signal. Spurious actuation in SS could be caused by the high containment pressure signal, the low pressurizer pressure signal, or by the low steamline pressure signal.

Once the ECCS is actuated, borated water is pumped from the refueling water storage, by the centrifugal charging pumps, into the cold leg of each RCS loop. The safety injection pumps are also started; but they cannot deliver any flow, since the shutoff head of these pumps is lower than the nominal RCS pressure.

As an ANS Condition II event, the inadvertent actuation of the ECCS, is expected to result, at worst, in a reactor trip with the plant being capable of returning to operation. The event must not propagate to cause a more serious fault (i.e., ANS Condition III or IV event), or result in fuel rod failures or RCS or secondary system overpressurization.

For this event, fuel rod failures are not likely to occur, due to the high pressures and low power levels that are present throughout the transient. RCS or secondary system overpressurization is not likely to occur, since heat removal requirements are very low (mainly decay heat) and the charging pump shutoff head is less than 110 % of the RCS design pressure. Therefore, the principal concern, when evaluating this event, is the possibility of propagation of the event to a more serious fault. This can occur if the event causes the pressurizer power-operated relief valves or safety valves to open when the pressurizer is water-solid. If these valves are not qualified for water relief, they must be assumed to stick open. If any of these valves fails to reseat properly, then the result would be a SBLOCA, an ANS Condition III event (i.e., an infrequent event).

For the inadvertent actuation of the ECCS, the licensee presents analyses which indicate that, although the pressurizer becomes water-solid, the operator prevents the discharge of water through any relief or safety valves by terminating the ECCS flow just before any of the valve opening set-pressures are attained, at 10.1 minutes. The staff noted that the operator was assumed to be terminating the ECCS flow at a time when the pressurizer was water-solid, and pressure was rising at a very rapid rate. Under these conditions, consideration of pressure measurement and setpoint uncertainties raise doubt as to whether the pressurizer power-operated relief valves could be relied upon to remain closed during the event. If any of the pressurizer power-operated relief valves were to open, even for a short time, while the pressurizer is water-solid, the valves would have to be assumed to stick open, which would create a SBLOCA, and thereby fail to meet the ANS Condition II acceptance criterion that prohibits the propagation of an ANS Condition II event into a more serious event.

The NRC staff informed the licensee that SS's analysis of the inadvertent actuation of the ECCS does not demonstrate that this ANS Condition II acceptance criterion would be satisfied. However, this acceptance criterion can be met, for example, by performing a licensing analysis which shows that there is adequate time for the operator to terminate the event before the pressurizer becomes water-solid, or that pressurizer power-operated relief valves are qualified, as a safety system, to mitigate this event.

The staff also informed the licensee it would accept an interim analysis, similar to that described in GL 91-18 [Reference 50], based upon an assumption of nominal plant conditions, which would demonstrate the event can be terminated by the operator before the pressurizer would become water-solid. The licensee is required to submit an acceptable licensing analysis, prior to startup following refueling outage 11, which demonstrates that the inadvertent actuation of the ECCS event would not develop into an ANS Condition III or Condition IV event, and that the other ANS Condition II acceptance criteria, regarding fuel damage and overpressurization, continue to be satisfied.

The licensee performed an interim analysis [Reference 11] in which nominal values were assumed for initial pressurizer pressure and level, and post-trip steam dumping to the condenser was assumed to be available. The results indicate that the pressurizer is predicted to fill in about 15 minutes. The licensee has also verified that the operator will terminate the ECCS flow within 10.1 minutes. Therefore, the pressurizer power-operated relief valves would not be expected to open and discharge water at any time during the event.

The staff agrees that the licensee's interim analysis of the inadvertent actuation of the ECCS demonstrates, with high confidence, that there is enough time before the pressurizer is predicted to become water-solid for the operator to terminate the transient, and thereby prevent any of the pressurizer relief or safety valves from opening and discharging water. The staff also agrees that the licensee has provided a reasonable assurance of safety for operation until the next refueling outage, by which time the licensee has committed to resolve this issue (e.g., by analysis or by plant modifications) which will assure that the plant will continue to meet the regulatory requirements following implementation of the proposed SPU with respect to the inadvertent actuation of the ECCS.

The licensee stated [Reference 11] the following commitment:

> Prior to startup from refueling outage 11, FPL Energy Seabrook commits to either upgrade the controls for the pressurizer power operated relief valves to safety-grade status and confirm the safety-grade status and water-qualified capability of the pressurizer power operated relief valves, pressurizer power operated relief valve block valves and associated piping or to provide a reanalysis of the inadvertent safety injection event, using NRC-approved methodology, that concludes that the pressurizer does not become water-solid within the minimum allowable and verifiable time for operators to terminate the event.

The NRC staff finds reasonable assurance that the event will not propagate to cause a more serious fault considering the interim analysis and the licensee's commitment as stated above and, therefore, is acceptable.

3.2.13.2.20 CVCS Malfunction that Increases RCS Inventory

The CVCS malfunction that increases RCS inventory is an ANS Condition II event that is evaluated for the effects of adding water inventory to the RCS. This event could be caused by operator error or a failure in the pressurizer level control system. In this case, the fault is assumed to be a spurious low pressurizer water level signal, demanding that charging flow increase to its maximum rate, with normal CVCS valve configuration. Makeup water, of a boron concentration that is equal to the boron concentration in the RCS, is added until the operator acts to terminate the flow. The analysis results indicate that pressurizer fills at about 13 minutes (this case is also mentioned in Section 3.2.13.2.16 of this SE). Therefore the operator has sufficient time to diagnose the problem and take corrective actions to terminate the event. Terminating the event, shortly after ten minutes, would prevent the pressurizer from becoming water-solid, and thereby preclude the possibility of a PORV opening, discharging water, and failing to reseat properly. A stuck-open PORV, under these circumstances, would not satisfy the ANS Condition II acceptance criterion which prohibits the escalation of an ANS Condition II event into a more serious event.

The staff reviewed the licensee's analyses of the CVCS malfunction transients and concludes that the licensee's analysis was performed using an acceptable analytical model. The staff finds the licensee demonstrated that the pressurizer would not become water solid before the operator can terminate the transient, by shutting off the charging flow, and thereby prevent this event from escalating to a more serious event. The staff concludes that SS will continue to meet the regulatory requirements following implementation of the proposed SPU with respect to the CVCS malfunction transient.

3.2.13.2.21 Inadvertent Opening of a Pressurizer Safety or Relief Valve

The inadvertent opening of a pressure relief valve results in a decrease in reactor coolant inventory decrease and RCS pressure. The inadvertent opening of a pressurizer PORV is classified as an ANS Condition II event, and the failure of a pressurizer safety valve is classified as an ANS Condition III event. A reactor trip normally occurs due to low RCS pressure or overtemperature ΔT. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.6.1.

Analysis of the accidental depressurization of the RCS, as an ANS Condition II event, would be postulated to occur as the result of an inadvertent opening of a pressurizer relief valve. Since a safety valve has about twice the steam relief capacity of a relief valve, its opening would cause a more rapid depressurization of the RCS. The applicant has conservatively evaluated the accidental depressurization of the RCS associated with an inadvertent opening of a pressurizer safety valve, an ANS Condition III event, while adhering to the more restrictive acceptance criteria of an ANS Condition II event.

The rapidly decreasing RCS pressure causes an erosion of thermal margin that could eventually lead to a demand for a reactor trip from the overtemperature ΔT RPS logic. In fact, this event is one of the events used to determine the calculated setpoints for the overtemperature ΔT trip signal. In some cases, the reactor would trip on low pressurizer pressure.

The accidental depressurization transient is analyzed with the NRC-approved RETRAN code [Reference 32]. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. The Code computes pertinent plant variables, including temperatures, pressures, and power level. This accident analysis is performed in accordance with the RTDP, in order to calculate the minimum DNBR during the transient. The licensee's analysis results indicate that the inadvertent opening of a pressurizer safety valve would not lead to a violation of the DNB design SAFDL. Therefore, no fuel damage is predicted, and the ANS Condition II acceptance criteria are satisfied.

The NRC staff has reviewed the licensee's analyses of the inadvertent opening of a pressurizer pressure relief valve event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to the inadvertent opening of a pressurizer pressure relief valve event.

3.2.13.2.22 <u>ATWS</u>

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in GDC-20. The regulation at 10 CFR 50.62 [Reference 34] requires that each PWR must have equipment that is diverse from the reactor trip system to automatically initiate the auxiliary (or emergency) FW system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must perform its function in a reliable manner and be independent from the existing reactor trip system.

The NRC staff's review was conducted to ensure that (1) the above requirements are met, and (2) the setpoints for the ATWS mitigating system actuation circuitry (AMSAC) remain valid for the proposed SPU. In addition, the NRC staff verified that the consequences of an ATWS are acceptable. The acceptance criterion is that the peak primary system pressure should not exceed the ASME Code Service Level C limit of 3200 psig (3215 psia). The peak ATWS pressure is primarily a function of the MTC and the primary system relief capacity. The NRC staff reviewed (1) the limiting event determination, (2) the sequence of events, (3) the analytical model and its applicability, (4) the values of parameters used in the analytical model, and (5) the results of the analyses. The NRC staff reviewed the licensee's justification of the applicability of generic vendor analyses to its plant and the operating conditions for the proposed SPU. Review guidance is provided in Matrix 8 of RS-001.

The final ATWS rule, 10 CFR 50.62(c) [Reference 34], requires that W-designed plants such as the SS install AMSAC circuitry to initiate a turbine trip and actuate emergency FW flow independent of the RPS. AMSAC has been installed at the SS, thus satisfying the requirements of 10 CFR 50.62.

The basis for the final ATWS rule and the AMSAC circuitry design is supported by W analyses documented in NS-TMA-2182 [Reference 35]. These analyses were performed using the NRC-approved LOFTRAN Code [Reference 33], and based on the guidelines published in NUREG-0460 (1978) [Reference 36]. Appendix A of WASH-1270 [Reference 23] states that in evaluating the RCS boundary for ATWS events, the calculated RCS transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the emergency conditions as defined in the ASME Nuclear Power Plant Components Code, Section III (Service Limit C).

Based on a review of RVs for 2-, 3- and 4-loop plants, W determined that the maximum allowable pressure for the RV is 3200 psig (or 3215 psia) and provided reference analyses [Reference 35] for 2-, 3-, and 4-loop plant designs with several different SG models. The ATWS events that produced the highest RCS overpressure transients were the loss-of-load and loss-of-FW events. The licensee analyzed these two events at the SPU conditions to ensure that the basis for the final ATWS rule continues to be met.

The results indicated that the highest RCS pressures generated by the ATWS events do not exceed the ASME Code Level C service limit stress criteria of 3200 psig (3215 psia). The highest RCS pressure attained, during the loss of load ATWS, was 3173 psia [Reference 5].

The staff agrees that SS continues to meet the analytical basis for the final ATWS rule for operation under SPU conditions.

The NRC staff has reviewed the information submitted by the licensee related to ATWS and concludes that the licensee has adequately accounted for the effects of the proposed SPU on ATWS. The NRC staff concludes that the licensee has demonstrated that the AMSAC will continue to meet the requirements of 10 CFR 50.62 following implementation of the proposed SPU. The licensee has shown that the plant is not required by 10 CFR 50.62 to have a diverse scram system. Additionally, the licensee has demonstrated, as explained above, that the peak primary system pressure following an ATWS event will remain below the acceptance limit of 3200 psig. Therefore, the NRC staff finds the proposed SPU acceptable with respect to ATWS.

3.2.13.2.23 Changes to TSs

TS 2.C.(1) Maximum Power Level

The licensee proposed to change the maximum core power level from 3411 MWt to 3587 MWt. The TS change reflects the actual proposed change in the plant and it is consistent with the results of the licensee's supporting safety analyses. The staff finds this proposed change acceptable.

TS 1.28 RTP

The licensee proposed to change the rated core power level from 3411 MWt to 3587 MWt. The TS change reflects the actual proposed change in the plant and it is consistent with the results of the licensee's supporting safety analyses. The staff finds this proposed change acceptable.

TS 2.1.1.1 SLs

The licensee proposed to revise the DNBR from 1.17 to 1.14 using WRB-2M DNB correlations. It is based on SPU parameters and it is acceptable.

TSs 4.2.2.2.g.1) and 4.2.2.2.g.2)

The proposed TS change expands the core region to which the limits specified in Specification 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f are applicable by 10%. Therefore, the proposed TS change is acceptable.

TS 3.2.5 DNB Parameters

The licensee proposed to revise the safety analysis flow value from 382,800 gpm to 374,400 gpm. Also, it will change the MMF value from 392,800 gpm to 383,800 gpm. It is based on the safety analyses and it is acceptable.

TS 3.7.1 MSSVs

The licensee proposed to revise the thermal power limits for maximum inoperable safety valves on any operating SG for LCO 3.7-1. When one or more SGs with one MSSV inoperable, the licensee proposed to reduce the RTP from 66% to 60%. When one or more SGs with two MSSVs inoperable, the licensee proposed to reduce RTP from 47% to 42%. When one or more SGs with three MSSVs inoperable, the licensee proposed to reduce RTP from 28% to 25%. The licensee will incorporate the proposed changes in Table 3.7-1 accordingly. The proposed changes to reduce the maximum allowable power levels listed in Table 3.7-1 reflect new limits corresponding to the SPU steam flow conditions. Each SG has five MSSVs for overpressure protection. A minimum of two operable safety valves per SG ensures that sufficient relieving capacity is available for the allowable thermal power restriction in Table 3.7-1. Therefore, this TS change is acceptable.

TS 6.8.1.6.b Administrative Controls

WCAP-12945-P-A is used for LOCA analysis for SPU conditions. In the docketed March 17, 2004 application, the licensee inadvertently omitted a marked-up TS page showing the change in item 1 of TS 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:” to include this WCAP-12945-P-A. Although the revision to item 1 to include this reference (i.e. WCAP-12945-P-A) was not docketed, the NRC staff considers that the list of analytical methods specified in TS 6.8.1.6.b. should include this reference. Including this revision will not change the NRC staff’s original proposed no significant hazards determination. Furthermore, the NRC staff reviewed and approved the use of WCAP-12945-P-A. Therefore, the NRC staff considers that it is acceptable for the licensee to make this revision.

The licensee agreed to revise item 1 by adding WCAP-12945-P-A, “Code Qualification Document for Best Estimate LOCA Analysis,” Volume 1, Rev. 2, and Volumes 2 through 5, Rev. 1, Bajorck, S.M., et al., 1998, which replaces WCAP-10266-P-A, Rev. 2 with Addenda (proprietary) and WCAP-11524-1, Rev. 2 with Addenda (non-proprietary), “The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code,” March 1987.

The licensee proposed to revise the list of the analytical methods specified in TS 6.8.1.6.b, used to determine the core operating limits. The licensee revised item 5 by adding WCAP-14565-P-A (proprietary), “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis”, October 1999, which replaces letter from T.H. Essig (NRC) to H. Sepp (W), “Acceptance for Referencing of Licensing Topical Report,” and WCAP-14565-P (proprietary), “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal Hydraulics Safety Analysis,” January 1999. It also adds, WCAP-15025-P-A, “Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17/17 Rod Bundles with Modified LPD Mixing Vane Grids,” April 1999.

WCAP-14565-P-A and WCAP-15025-P-A are used for non-LOCA analyses and DNBR analyses for SPU conditions, respectively, and these changes are acceptable.

In item 6, the licensee added, WCAP-8745-P-A, “Design Basis For the Thermal Overpower ΔT and Thermal Overtemperature ΔT trip functions,” September 1986.

In item 7, the licensee replaced WCAP-14551-P (proprietary), “Westinghouse Setpoint Methodology for Protection Systems, SS Nuclear Power Station Unit 1, 24 Month Fuel Cycle Evaluation,” June 1998. WCAP-14551-P was specifically issued for the 24-month fuel cycle program for SS which was never implemented and submitted for NRC review. Therefore, it is acceptable to remove from the TS.

In Item 14, the licensee removed WCAP-8385-P (Proprietary), “Power Distribution Control and Load Following Procedures,” September 1974. The staff considers this change acceptable since it is no longer applicable.

3.2.14. Summary and Conclusions

The staff reviewed the licensee's evaluations, analyses and proposed TS changes to support operation of SS at the proposed SPU level of 3587 MWt. Based on its review, the staff finds that the supporting safety analyses were performed with NRC-approved computer codes and methods; the input parameters of the analysis adequately represent the plant conditions at the power level assumed in each analysis; and the analytical results are within the applicable acceptance criteria. Therefore, the staff concludes that the supporting analyses are acceptable. The staff also finds that the proposed TS changes discussed in this evaluation adequately reflect the results of the acceptable supporting analysis, and therefore, concludes that the proposed TS changes are acceptable for the implementation of the SPU for the SS.

3.3 Electrical Systems

3.3.1 Environmental Qualification (EQ) of Electrical Equipment

3.3.1.1 Regulatory Evaluation

The term "environmental qualification" applies to equipment important to safety to assure this equipment remains functional during and following design-basis events. The NRC staff's review covers the environmental conditions that could affect the design and safety functions of electrical equipment including I&C. The NRC staff's review is to ensure compliance with the acceptance criteria, thus ensuring that the equipment continues to be capable of performing its design safety functions under all normal environmental conditions, AOOs, and accident and post-accident environmental conditions. Acceptance criteria are based on 10 CFR 50.49 as it relates to specific requirements regarding the qualification of electrical equipment important to safety that is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

3.3.1.2 Technical Evaluation

Review was performed for the new accident temperature, pressure, humidity, submergence and radiation dose associated with the uprate environmental conditions in the EQ program. Evaluation of the EQ of equipment demonstrates that the equipment will remain qualified under SPU conditions, but with minor changes in the environments the equipment will experience at SPU conditions. Specifically, in the qualification evaluation process, the changes to equipment environments resulting from AOOs, normal operation, off-normal operation, accident and post-accident conditions are reviewed. Comparisons were made between the environmental conditions to which the equipment is currently qualified and the environmental conditions that will be present following the implementation of the SPU. Where there is a qualification challenge, further evaluations specific for the equipment were performed. The evaluations demonstrated the continued qualification of the equipment, or identified the requirement for equipment upgrades or changes by ensuring that the margins required by Institute of Electrical and Electronics Engineers (IEEE) 323-1974 and the Equipment Qualification Program are maintained.

3.3.1.2.1 Inside Containment

Normal service conditions and operational occurrences for radiation, pressure, temperature and humidity do not change. The SPU analysis of a DBA demonstrates that the equipment qualification temperature profile bounds the SPU accident profile. The pressure profile for the equipment qualification bounds the SPU pressure results. The radiation doses inside containment are increased due to SPU. Revised total integrated doses that combine normal operation service conditions with the DBA have been compared to the original qualification value. The results of the comparison show that the equipment continues to be qualified at the SPU conditions. Other parameters that affect the qualification of equipment are humidity, submergence and chemical spray. Humidity and submergence do not change. The other DBA conditions, such as peak temperature, peak pressure, and the chemical environments will remain bounding as a result of the SPU.

3.3.1.2.2 Outside Containment

Normal service conditions and operational occurrences for temperature and humidity do not change following implementation of the SPU. The accident environments outside containment have been evaluated. There are no changes to the qualification of equipment as a result of the high-energy line breaks (HELBs) in this area. For areas outside containment with harsh radiation environments, the equipment total integrated doses have been scaled up, where applicable, to account for increases in normal operating and post-accident radiation levels as a result of the SPU. The results of the comparison show that the electrical equipment continues to be qualified at SPU conditions.

3.3.1.3 Summary

The NRC staff has reviewed the licensee's submittal of the effects of the proposed power uprate on the EQ of the electrical equipment and concludes that the electrical equipment is capable of performing its safety function under the environmental conditions that could result from DBAs in accordance with 10 CFR 50.49, and is acceptable.

3.3.2 Offsite Power System

3.3.2.1 Regulatory Evaluation

The offsite power system includes two or more physically-independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covers the information, analyses and documents for the offsite power system and the stability studies for the electrical transmission grid. The focus of the review relates to the basic requirement that loss of the nuclear unit, the largest operating unit on the grid or the most critical transmission line will not result in the LOOP to the plant. Branch Technical Position (BTP) Instrumentation & Control Systems Branch (ICSB) 11, "Stability of Offsite Power Systems," outlines an acceptable approach to addressing the issue of stability of offsite power systems. Acceptance criteria are based on GDC-17 of Appendix A to 10 CFR Part 50. Specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to 8.2 and BTP PSB-1 and ICSB-11.

3.3.2.2 Technical Evaluation

The main generator is rated 1350 megavolts-amperes (MVA), 25 KV, 60 HZ, 0.92 power factor, 1800 rpm at 75 psig hydrogen pressure, with a water-cooled stator. The output of the main generator is delivered to the low voltage windings of three single-phase generator step-up (GSU) transformers via the main isolated phase bus (IPB) duct. An IPB tap bus connects the main generator output to the unit auxiliary transformers (UATs). The generator circuit breaker is located between the generator output terminals and the IPB tap to the GSU transformers and UATs. The GSU transformers and UATs form part of the immediate access circuit from the preferred (offsite) power supply to the onsite distribution system when the generator circuit breaker is open. Offsite power from the 345 KV switchyard is supplied to the 13.8 KV and 4.16 KV buses via the reserve auxiliary transformers (RATs) during plant start-up, outage, and DBA conditions. The electrical distribution system has been previously evaluated to conform to GDC-17.

3.3.2.2.1 Grid Stability

An uprate system impact study was completed to evaluate the system impacts in accordance with New England Power Pool (NEPOOL) Reliability Standards. It compared performance of the system before and after the proposed SPU to demonstrate the impact under a prescribed set of initial conditions and contingencies established in cooperation with the NEPOOL transmission owners and Independent System Operator-Now England (ISO-NE). The evaluation considered an analyzed electrical output of 1295 megawatts electric (MWe), which is the maximum capability of the existing generator without modifications and is 86 MWe above the present gross output of 1209 MWe. The study demonstrated system performance with and without SPU for pre-contingency and post-contingency voltages and line loading, and for dynamic response to system disturbances. The study indicated that the SPU of SS will not have a significant adverse effect on the reliability or operating characteristics of the station or on the offsite electrical system. The staff was concerned regarding the depletion of the megavolt-ampere reactive (MVAR) with the power uprate and asked the licensee, in an RAI dated August 18, 2004, to address the following points:

1. Provide details about the grid stability analysis including assumptions and results and conclusions for the power uprated condition.

2. a. Identify the nature and quantity of MVAR support necessary to maintain post-trip loads and minimum voltage levels.

   b. Identify what MVAR contributions the SS Unit No. 1 is credited for providing to the offsite power system or grid.

   c. After the power uprate, identify any changes in MVAR quantities associated with Items a. and b. above.

   d. Discuss any compensatory measures to adjust for any shortfalls in Item c. above.

e. Evaluate the impact of any MVAR shortfall listed in Item d. above on the ability of the offsite power system to maintain minimum post-trip voltage levels and to supply power to safety buses during peak electrical demand periods. The subject evaluation should document any information exchanges with the transmission system operator.

In its response dated August 25, 2004, the licensee stated that the SPU will not change the generator capability curve. The lagging reactive capability will be reduced from 560 MVAR lagging to 367 MVAR lagging after the SPU. In addition, the SPU will result in the leading reactive capability being changed from 75 MVAR leading to 0 MVAR leading. For the SPU the generator reactive capability will be sufficient to meet both pre-contingency and post-contingency voltage criteria. A total of 364 MVAR lagging of the generator reactive capability is needed to maintain the SS 345 kV voltage at a level of 1.035 per unit in the pre-contingency state for the peak load level. As long as the SS reactive capability after the SPU is more than 364 MVAR lagging, shunt capacitance does not need to be added at this phase. The SPU has no significant adverse impact on thermal or voltage performance for the system conditions and contingencies that were studied.

Also, the SPU has no significant impact on the stability performance of the system for the conditions and contingencies that were studied except the following:

The SS unit, with implementation of the proposed 1295 gross MWe SPU will be required by the licensee's operating procedure to limit the gross output level in real-time operation such that the net loss of the source that results from a contingent SS generator trip is at or below the real-time-based maximum allowable net source loss for the NEPOOL control area. Any reductions to the gross output of SS to meet this requirement will be required within 30 minutes of being directed to do so by ISO-NE.

The steady state and dynamic performance of the SS SPU are acceptable as studied without any remedial measures except as noted above. SS's design will continue to meet GDC-17 at SPU conditions and is acceptable.

3.3.2.2.2 Main Generator

The main generator is rated 1350 MVA, 25 KV, 60 HZ, 0.92 power factor, 1800 rpm at 75 psig hydrogen pressure, with a water-cooled stator. The main generator real power output of 1295 MWe bounds the expected generator output for the proposed licensed core power of 3587 MWt, which is calculated based on heat balance model calculations. The evaluation of the main generator was based upon a comparison between the generator capability curve and the anticipated operating requirements when the machine operates at SPU conditions. The generator capability curve shows that the machine is capable of continuous operation at an output of 1242 MWe (0.92 lagging power factor) up to and including 1350 MWe (unity power factor) at 75 psig hydrogen pressure. The maximum required generator power output is 1295 MWe. Therefore, the real power output capability of the main generator is greater than the real power output required at SPU. The reactive power capability of the main generator from the generator capability curve is approximately 367 MVAR lagging, when the unit operates at 1295 MWe. The generator operation at the specified values corresponds to a generator lagging power factor of 0.960 at SPU conditions and 75 psig hydrogen pressure.

The reactive capability of the main generator satisfies the SS reactive power commitments to ISO-NE of 367 MVAR lagging and 0 MVAR leading.

The NRC staff reviewed the licensee's submittal and concludes that the plant will continue to operate the main generator within its design rating at the SPU and it will maintain proper voltages for the operation of plant equipment and, therefore, the design is acceptable.

3.3.2.2.3 GSU Transformers

The GSU transformer's maximum design rating is 1380 MVA at 65 EC rise. Evaluation of the GSU transformers was based upon a comparison between the transformer design ratings and the anticipated maximum transformer loading requirements when the unit operates at SPU conditions. A load flow/voltage profile system analysis was used to calculate the 1272 MVA maximum load on the GSU transformers with station auxiliaries supplied from the UATs consistent with the normal plant configuration when the unit is operating at full power. A load flow/voltage profile system analysis was used to calculate the 1332 MVA maximum load on the GSU transformers with station auxiliaries supplied from the RATs when the unit is at full power which is below the maximum design rating of 1380 MVA at 65 EC rise.

The NRC staff reviewed the licensee's submittal and concludes that the GSU transformers will operate at its rated loading and, therefore, operating the GSU transformers at the SPU is acceptable.

3.3.2.2.4. IPB Duct

The design rating of the IPB duct main bus is 35 kA forced-cooled. The design rating of the IPB duct, tap bus to GSU transformers is 20 kA self-cooled. The design rating of the IPB duct, extension tap bus and tap bus to UAT is 3 kA self-cooled. Evaluation of the IPB main and tap buses was based upon a comparison between the maximum anticipated full-load current at SPU and the design ratings of the main and tap bus conductors. This evaluation is based on station auxiliaries being fed from the UATs, since this results in the worst-case IPB loading. The evaluation results demonstrate that the loadings of the IPB main (33.1 kA) tap bus to a GSU transformer (18.25 kA), and tap bus to UAT (0.98 kA) are below the design rating at SPU conditions.

The NRC staff reviewed the licensee's submittal and concluded that the IPB duct will operate at its rated loading and, therefore, operating the IPB at the SPU is acceptable.

3.3.2.2.5 UAT

Each UAT is a three-winding transformer and its maximum design rating is 40.32 MVA at 65 EC. The existing loading is 39.11 MVA. Evaluation of the UATs was based upon a comparison between the transformer nameplate design ratings and the anticipated maximum transformer loading requirements when the unit operates at SPU conditions. To determine the impact of SPU conditions on the UATs, a baseline for transformer loading was developed that represents the existing loading. Load changes due to SPU conditions were added to the baseline using a computer model. The total transformer load is 40.20 MVA which is less than the maximum design rating of 40.32 MVA at 65 EC rise. The evaluation confirms that the existing UAT design ratings are adequate to support unit operation at SPU conditions.

The NRC staff reviewed the licensee's submittal and concluded that the UATs will operate at their rated loading and, therefore, operating the UATs at the SPU is acceptable.

3.3.2.2.6 Reserve Auxiliary Transformer (RAT)

Each RAT is a three-winding transformer and is rated at 40.32 MVA at 65 EC. Evaluation of the RATs was based upon a comparison between the transformer nameplate design ratings and the anticipated maximum transformer loading requirements when the unit operates at SPU conditions. To determine the impact of SPU conditions on the RAT, a baseline for transformer loading was developed that represents the existing loading. Load changes due to SPU conditions were added to the baseline using a computer model. The total transformer load is 40.28 MVA which is less than the maximum design rating of 40.32 MVA at 65 EC rise. The evaluation confirms that the existing RAT design ratings are adequate to support unit operation at SPU conditions.

The NRC staff reviewed the licensee's submittal and concluded that the RATs will operate at their rated loading and, therefore, operating the RATs at the SPU is acceptable.

3.3.2.2.7 Generator Circuit Breaker

The generator circuit breaker is rated at 35 kA. Its evaluation was based upon a comparison between the maximum anticipated full-load current at SPU conditions and the design rating of the circuit breaker with the generator operating at maximum gross output. The results demonstrate that the circuit breaker rating of 35 kA exceeds the anticipated worse-case loading at SPU conditions.

The NRC staff reviewed the licensee's submittal and concluded that the generator circuit breaker will operate at its rated loading and, therefore, operating the generator circuit breaker at the SPU is acceptable.

3.3.2.2.8 ac Distribution System

The calculated worst-case current values for each 13.8 kV or 4.16 kV switchgear bus incoming breaker during a maximum full load at SPU conditions is bounded by the equipment design ratings. Motor terminal voltage values for starting or running reactor coolant pump, condensate pump and heater drain pump motor drives at SPU conditions, during steady-state maximum full load conditions, are above the minimum required voltage; and motor terminal voltage values for the other running motors (on the same bus) are above the minimum required voltage. Brake horsepower requirements for the condensate pump and heater drain pump motor drives at SPU conditions are determined by the licensee to be within motor nameplate ratings. The calculated full-load current values for reactor coolant pump, condensate pump and heater drain pump motor drives at SPU conditions do not exceed the derated ampacities of motor feeder cable and electrical penetrations. Therefore, the motor feeder cable and electrical penetration requirements at SPU conditions are bounded by the existing design ratings.

The NRC staff reviewed the licensee's submittal and concluded that the proposed SPU does not have any adverse impact on the ac distribution system including the switchgear bus, circuit breakers, and motors, and they are bounded by the equipment design ratings and the design is, therefore, acceptable.

3.3.2.2.9 Summary

The anticipated operating requirements associated with SPU conditions are bounded by the design ratings of the main generator, IPB, generator circuit breaker, GSU transformers, UATs and RATs. There is no change needed to the equipment functional design and the equipment continues to meet the requirements of GDC-17 following implementation of the proposed SPU. The existing physical and electrical separation is not affected, and the power block equipment has the capacity and capability to supply power to the safety-related loads and other required equipment at SPU conditions and the design is, therefore, acceptable.

3.3.3 EDGs

3.3.3.1 Regulatory Evaluation

The ac onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to the safety-related equipment. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the ac onsite power system. Acceptance criteria are based on GDC-17 as it relates to the capability of the ac onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

3.3.3.2 Technical Evaluation

It was determined that there are no load additions or modifications to the present EDG loading due to the SPU. Therefore, there is no impact to the existing EDG loading analysis, which is bounding at SPU conditions. The EDGs are capable of performing their design and licensing functions at SPU conditions. Since the EDG loading is bounded by existing analysis, the existing protective relay settings are not impacted by SPU conditions.

3.3.3.3 Summary

The NRC staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system's functional design. The NRC staff further concludes that since there are no changes to the EDG loading conditions, the ac onsite power system will continue to meet the requirements of GDC-17 following implementation of SPU. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the onsite ac power system.

3.3.4 Onsite Direct Current (dc) Power Systems

3.3.4.1 Regulatory Evaluation

The dc power systems include those dc power sources and their distribution systems and auxiliary supporting systems provided to supply motive or control power to safety-related equipment. The NRC staff's review covered the information, analyses, and referenced documents for the dc onsite power system. Acceptance criteria are based on GDC-17 and 10 CFR 50.63 as they relate to the capability of the onsite electrical power to facilitate the functioning of structures, systems, and components (SSCs) important to safety. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2.

3.3.4.2 Technical Evaluation

The licensee evaluated the dc systems and determined that no new dc loads were added and no dc load increases were identified for the existing loads. In addition, the station blackout (SBO) and Appendix R program evaluations did not result in any 125-volts-dc system load changes. The dc system is not affected by SPU and continues to have the capacity and capability to supply power to the safety loads as required in GDC-17.

3.3.4.3 Summary

The staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the dc onsite power system and concludes that the SPU does not affect the dc system. The staff further concludes that the dc onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed power uprate. Therefore, the staff finds the proposed power uprate acceptable with respect to the dc onsite power system.

3.3.5 SBO

3.3.5.1 Regulatory Evaluation

An SBO refers to the complete LOAC power to the essential and nonessential switchgear busses in a nuclear power plant. SBO involves the LOOP concurrent with turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from Aac. The NRC staff's review focused on the impact of the proposed power uprate on the plant's ability to cope with and recover from an SBO event. The specified blackout duration is based on the factors detailed in 10 CFR 50.63. Specific review criteria are contained in SRP Section 8.1 and Appendix B to SRP 8.2.

3.3.5.2 Technical Evaluation

SS SBO coping duration is 4 hours. The SPU does not impact the above offsite power design characteristics, modify the emergency ac power system configuration or affect the EDG reliability. Therefore, the SPU has no impact on SBO coping duration. The current CST inventory requirement for decay heat removal during the four-hour SBO coping period is 131,137 gallons. This inventory requirement will increase slightly to 137,000 gallons, resulting from increased decay heat related to the SPU. This value includes decay heat removal,

removal of sensible heat and SG level shrinkage. This requirement is below the minimum CST inventory of 196,000 gallons. Therefore, the current CST minimum inventory supports SBO decay heat removal requirements for the SPU. The SPU does not increase the loads required to cope with the SBO, nor does it impact the load-shedding requirement. SBO battery sizing requirements are not impacted by the SPU. Therefore, the SPU has no impact on SBO battery capacity. The only air-operated valves (AOVs) required for decay heat removal during the 4-hour SBO coping period are the atmospheric steam dump valves. The backup air supply is completely independent of ac power and is sized for atmospheric steam dump valve operation over a 10-hour period. Therefore, SBO compressed air requirements will be met for SPU conditions. The areas containing equipment required to cope with an SBO were evaluated for the effects of loss of ventilation at SPU conditions. Motor horsepower, electrical loading and fluid temperatures for systems utilized during the SBO 4-hour coping period are not affected by the SPU. Therefore, the SPU has no impact on the effects of loss of ventilation for SBO. The SPU has no impact on the requirements for containment isolation during SBO. The expected rates of reactor coolant inventory loss under SBO conditions will not increase as a result of the SPU. Therefore, makeup systems will not be required during SBO at SPU conditions to maintain core cooling under natural circulation conditions.

3.3.5.3 Summary

The staff has reviewed the licensee's submittal on the effect of the proposed power uprate on the plant's ability to cope with and recover from an SBO event for the period of time established on the plant's licensing basis. The staff concludes that the licensee has adequately evaluated the effects of the proposed power uprate on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following the implementation of the proposed power uprate. Therefore, the staff finds the proposed power uprate acceptable with respect to SBO requirements.

3.3.6 Conclusion

The NRC staff has reviewed the licensee's safety analyses of the impact of the proposed SPU on EQ of electrical equipment, grid stability, including performance of the main generator, main transformer, isophase bus, RATs, and UATs, EDG, and SBO. Results of these evaluations show that the increase in a core thermal power would have negligible impact on the grid stability, SBO, or the EQ of electrical components. This is consistent with GDC-17 of Appendix A to 10 CFR 50, 10 CFR 50.63, and 10 CFR 50.49 and the proposed change is, therefore, acceptable.

3.4 Civil and Engineering Mechanics

3.4.1 Regulatory Evaluation

This technical evaluation includes the structural and functional integrity of piping systems, components, and their supports, including core support structures (CSS), which are designed in accordance with the rules of the ASME Code, Section III, Division 1, American National Standards Institute (ANSI) B31.1 Power Piping Code, and GDC-1, 2, 4, 10, 14, and 15. The staff review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of piping systems, components, component internals, and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes.

The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR Part 50, 50.55a and GDC-1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC-2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4 as it relates to structures and components important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions; (4) GDC-10 as it relates to reactor internals being designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (5) GDC-14 as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and (6) GDC-15 as it relates to the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded.

The specific review areas are contained in the NRC SRP Section 3.9. The review also includes the plant specific provisions of Generic Letter (GL) 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” and GL 96-05, as related to plant-specific program for motor operated valves, GL 95-07, “Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves,” as related to the pressure locking and thermal binding for safety-related gate valves, and the plant specific evaluation of the plant’s program for GL 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,” regarding the over-pressurization of isolated piping segments.

3.4.2 <u>Technical Evaluation</u>

The NRC staff reviewed the SS power uprate amendment request, as it relates to the effects of the power uprate on the structural and pressure boundary integrity of the NSSS and balance-of-plant (BOP) systems. Affected components in these systems include piping, in-line equipment and pipe supports, the RPV, CSS, RVIs, SGs, CRDMs, RCPs and pressurizer. The NRC staff’s evaluation concerning the effects of the power uprate on the pertinent components is provided below.

3.4.2.1 <u>RPV</u>

The proposed power uprate will increase the core power by approximately 5.2% above the currently authorized power level of 3411 MWt. The licensee reported that the power increase will result in changing the design parameters to those indicated in Table 2.3-1 [Reference 1].

The licensee evaluated the RV for the effects of the revised design conditions. The evaluation was performed for the limiting vessel locations with regard to stresses and fatigue cumulative usage factors (CUFs) in each of the regions, as identified in the RV stress reports for the core power uprated conditions. The regions of the RPV affected by the power uprate include outlet and inlet nozzles, CRDM housing, core support lugs, external supports, and the instrumentation tubes. In its amendment request, the licensee indicated that the evaluation of the RV was performed in accordance with the ASME Code, Section III, 1971 Edition with Addenda through the winter 1972, which is the code of record. Table 5.1-1 of Reference 1 provides the calculated maximum stresses and CUFs for the RV critical locations. The results indicate that the maximum primary plus secondary stresses are within the ASME Code allowable limits, and the CUFs remain below the allowable ASME Code limit of 1.0, with the exception of the outlet nozzle safe end and the bottom head instrumentation tubes. For these locations, the licensee performed simplified elastic-plastic analyses in accordance with Subsection NB of the ASME Code. In Reference [5], FPLE provided additional information on these analyses. The staff reviewed the information in Reference [2] and concurs with the licensee's conclusions that these locations meet ASME Code requirements for limiting stresses. Therefore, the staff agrees with the licensee's conclusion that the RPV continues to be in compliance with the ASME Code of record for the proposed power uprate conditions.

3.4.2.2 Reactor Core Support Structures and Vessel Internals

The licensee evaluated the core support structures and RVI for the SPU conditions. The evaluation included the lower core plate, lower support columns, lower core plate, core barrel, baffle-barrel region components, baffle-barrel bolts, upper core support plate, upper support columns, and upper core plate. The licensee indicated that the reactor internal components were not licensed to the ASME Code; however, the design of the reactor internals was evaluated in accordance with Subsection NG of Section III of the ASME Code.

The licensee evaluated these critical reactor internal components considering the SS SPU design conditions provided in Table 2.3-1 of Reference [1]. The licensee indicated that the calculated stress for the limiting reactor internals are within the ASME Code allowable limits. In Tables RAI 24-1 and RAI 24-2 of Reference [5], the licensee provided the calculated stresses and CUFs for the RVI, demonstrating that they are less than the ASME Code allowable limits. In addition, Reference 1 states that the licensee evaluated the flow induced vibration, which was found to remain within the allowable limits for the proposed power uprate condition. Based on the above evaluations, the staff agrees with the licensee's conclusion that the reactor internal components at SS will be structurally adequate for the proposed power uprate.

3.4.2.3 CRDMs

The pressure boundary portion of the CRDMs are those exposed to the vessel/core inlet fluid. SS has the L-106A1 CRDMs, full-length mechanisms manufactured by W. The CRDMs were designed to Section III of the ASME Code, 1974 Edition, summer 1974 Addenda, Sections NA and NB. The licensee evaluated the adequacy of the CRDMs by comparing the original generic evaluation for L-106A1 CRDMs against the design input parameters in Table 2.3-1 of Reference [1], and the revised transients in Section 3.0 of Reference [1]. In Reference [5], the licensee provided additional information demonstrating that the maximum stresses are below the ASME Code allowable limits for the SS SPU conditions. On the basis of its review, the staff concurs with the licensee's conclusion that the CRDMs continue to be in compliance with the ASME Code for the proposed power uprate.

3.4.2.4 SGs

The licensee reviewed the W Model F SGs at SS for the new design and transient conditions for SPU. The evaluation considered the range of RCS temperatures, tube plugging up to 10%, and a range of FW temperatures. The evaluation included moisture carryover, local tube wall dryout, structural integrity, tube vibration (flow-induced), and wear. The licensee concluded that the Model F SGs will support operation at SPU conditions.

The structural analysis was performed in accordance with the ASME B&PV Code, Section III, 1971 Edition, 1973 Addendum, which is the code of record. The maximum stresses, along with the ASME Code limits, are in Table 5.7.2-1 of Reference [1]. This table indicates that the primary-plus-secondary stresses exceed 3Sm (code-allowable stress limit) for several locations, and that simplified elastic plastic analyses were used to demonstrate compliance with the ASME Code. In Reference [5], the licensee provided additional details of the elastic-plastic analyses. The staff reviewed the information in Reference [5] and concurs with the licensee's assessment that these locations meet the ASME Code requirements.

The licensee evaluated flow-induced vibration and determined that the stability ratio increased for the SPU conditions, but remains below the acceptance limit of 1.0. In addition, the maximum fluid-induced displacements due to turbulence and vortex shedding are insignificant. As a result, the licensee concluded that the flow-induced vibration of SG tubes will remain within the allowable limits for the power uprate. The staff concurs with the licensee's conclusion.

On the basis of its review, the staff concludes that the licensee has demonstrated the maximum stresses and CUFs for the limiting SG components to be within the ASME Code allowable limits and are, therefore, acceptable for the proposed power uprate.

3.4.2.5 Reactor Coolant Pumps

The licensee reviewed the existing Model 93A-1 reactor coolant pumps for operating conditions and design transients associated with the SPU. The SS reactor coolant pumps were designed to Section III of the ASME Code, 1971 Edition with addenda through summer 1973 Addenda.

For SPU, the RCS pressure remains unchanged and the reactor coolant pump temperature decreases from the original design values. Due to lower allowable stresses at higher temperatures, higher temperatures are more limiting in the analyses; therefore, the current reactor coolant pump design is conservative for SPU normal operating conditions. For the transients, the current design analyses bound the transient temperature profiles for normal and upset conditions; therefore, there is no change due to SPU. In addition, there was no impact on ASME Code evaluations for the emergency conditions. The original qualification of the reactor coolant pumps was based on a fatigue waiver, in accordance with Section NB-3222.4(d) of the ASME Code, for which the reactor coolant pumps still qualify. However, fatigue analyses were performed as part of the elastic-plastic analyses, in accordance with Section NB-3228.3 of the ASME Code, for locations where the primary-plus-secondary stress exceeded 3Sm. In Reference [5], the licensee provided additional information on the fatigue waiver and the elastic-plastic analyses. The staff reviewed Reference [5] and concurs with the licensee's conclusions that the reactor coolant pumps meet ASME Code requirements for SPU conditions.

3.4.2.6 Pressurizer

The licensee evaluated the limiting design locations of the pressurizer components. The components in the lower end of the pressurizer (such as the surge nozzle, lower head well and penetration, and support skirt) are affected by the pressure and the hot leg temperature. The components in the upper end of the pressurizer (such as the spray nozzle, instrument nozzle, safety and relief nozzle, and upper head and shell) are affected by the pressure and the cold leg temperature for operation at the uprated conditions. The pressurizer was designed to the ASME Code, Section III, 1971 Edition, with addenda through summer 1973 Addenda.

The licensee determined that the parameters used in the existing design report bound the SPU conditions; therefore, the current design analysis remains the analysis of record. However, the licensee also stated that the design report shows a CUF close to the allowable limit of 1.0 for the surge nozzle, and that the original design did not include an evaluation of the effects of thermal stratification on the surge line. The licensee stated [Reference 5] that an additional evaluation of thermal stratification was performed for Millstone Power Station, Unit No. 3, which utilizes the same pressurizer model (including critical dimensions, materials, and ASME Code of record) as SS. That detailed evaluation removed excessive conservatism from the original design basis and demonstrated a significantly lower CUF (approximately 0.3) for the surge nozzle. The licensee concludes, based on the comparative analysis, that the CUF would remain below 1.0 even if thermal stratification were considered. As a result, the licensee concluded that the existing pressurizer components will remain adequate for plant operation at the proposed power increase. On the basis of its review, the staff agrees with the licensee's conclusion.

3.4.2.7 NSSS Piping and Piping Supports

The proposed power uprate involves an increase in temperature difference across the core and revisions to the transient evaluations. The licensee evaluated the NSSS piping and supports by reviewing the design basis analyses against the uprated power design system parameters, transients, loss-of-coolant-accident (LOCA) dynamic loads, and FW and main steamline breaks (MSLBs). The evaluation was performed for the RCL piping, primary equipment nozzles, primary equipment supports, pressurizer surge line piping, RCL branch nozzle loads, and Class 1 auxiliary piping.

The RCL piping and Class 1 auxiliary lines were designed to the requirements of the ASME Code, Section III, 1977 Edition with addenda through the summer 1979 Addenda. The RCL equipment supports were designed to the requirements of the ASME Code, Section III, Subsection NF, 1974 Edition, summer 1974 Addenda. The surge line thermal stratification was addressed in accordance with NRC Bulletin 88-11, with CUFs evaluated based on the requirements of the ASME Code, Section III, 1986 Edition. These are the codes of record for SS.

The licensee indicated that the proposed power uprate does not change the maximum NSSS pressure or the system deadweight and seismic loads. The SPU results in a small increase in RCL temperatures (in some design cases), which has an insignificant impact on the analyses of the RCL piping and components. Similarly, the impact due to the changes in the transients was small, with no significant effect on the RCL piping, branch piping, or Class 1 auxiliary piping. The controlling transients for the pressurizer surge line fatigue evaluation were not effected by SPU; therefore, the surge line fatigue evaluation remains bounding. For the LOCA hydraulic forcing functions, the licensee applied leak-before-break, consistent with its current licensing basis, and evaluated breaks in the 12-inch RHR lines, 14-inch pressurizer surge line, 10-inch accumulator lines, and secondary side breaks in the main steamline and FW line. The calculated stresses and CUFs are provided in Table 5.5-1 of Reference [1]. The maximum calculated stresses and CUFs are less than the Code ASME allowable limits.

On the basis of its review of the licensee's submittal, the staff concurs with the licensee's conclusion that the existing NSSS piping and supports, the primary equipment nozzles, and the auxiliary lines connecting to the primary loop piping will remain in compliance with the requirements of the ASME Code of record for SS and are, therefore, acceptable for the proposed power uprate.

3.4.2.8 BOP Systems and Motor-Operated-Valves (MOVs)

The licensee evaluated the adequacy of the BOP systems, structures, and plant programs for SPU conditions. For the CVCS, RHR system, and ECCS components, the licensee found that the maximum operating temperatures and pressures for SPU conditions are bounded by the system design. For the MSS, the SPU operating pressures and flow rates remain below the SS design limits. Also, the main steam isolation valve design conditions remain bounded by the SPU conditions. SPU does not change the limiting flow rate, which results from an MSLB from hot standby. The moisture separators/reheaters are being modified to improve performance and support SPU conditions. The condensate and FW system flows and pressures are within the design limits, and increases in flow rate will be factored into the flow-accelerated corrosion programs. The SG blowdown system (SGBS) flow rate does not change, and the system pressure and temperatures associated with SPU are bounded by the design of this system. For the cooling water systems (ultimate heat sink, service water, and CCW) the heat loads for SPU normal operation and post-LOCA remain within the system design heat loads.

The licensee evaluated the BOP piping in accordance with the ASME Codes of record, which are ASME Code, Section III, 1977 Edition, for safety-related piping and ANSI B31.1, 1973 Edition for non-safety-related piping. Table 8.5-1 of Reference [1] shows that the stresses for those piping systems that required reanalyses remain below the allowable limits of the applicable design codes.

The licensee evaluated the safety-related power-operated valves, including MOVs, AOVs, and solenoid-operated valves (SOVs), considering GL 89-10 and GL 95-07. There were no changes to the design basis temperature or pressure and no increases in worst-case differential pressure, since the values assumed in the analyses bound the SPU conditions. The valves evaluated in response to GL 95-07 will not be adversely affected by potential pressure locking and thermal binding because the SPU conditions are bounded by the existing program assumptions. Therefore, the licensee determined that the existing design bases for the MOVs, AOVs, and SOVs is acceptable for SPU conditions. On the basis of the above review, the staff concurs with the licensee's conclusions that the power uprate will have no adverse effects on the power operated valves at SS.

The licensee evaluated SS with respect to GL 96-06 regarding over-pressurization of isolated piping segments. SPU does not result in an increase in containment design temperature or pressure; therefore, the current evaluations for GL 96-06 remain bounding for SS SPU conditions.

As a result of the above evaluation, the staff agrees with the licensee's conclusion that the BOP piping, pipe supports and equipment nozzles, and valves remain acceptable and continue to satisfy the design-basis requirements for the proposed power uprate.

3.4.3 Summary

On the basis of its review in Section 3.4 of this SE, the staff concurs with the evaluations performed by the licensee for the NSSS and BOP piping, components, and supports, the RV and internal components, the CRDMs, SGs, reactor coolant pumps and the pressurizer. The staff finds the licensee's evaluation to be bounded by the licensing codes of record and the original design bases, and therefore, concludes the foregoing components to be acceptable for SS at the proposed core power level of 3587 MWt.

3.5 Dose Consequences Analysis

The licensee's letter of March 17, 2004 [Reference 1] indicated that the 5.2% (3587 MWt) SPU application relied upon the approval of the Alternate Source Term (AST) application that was submitted by the licensee by letter dated October 6, 2003 (Accession Number ML032890198). The AST LAR was approved by letter dated February 24, 2005 (Accession Number ML050320373). Through reanalysis of the radiological consequences of the UFSAR Chapter 15 accidents, the licensee proposed to revise the SS licensing basis to implement the AST. The inventory of fission products in the core and the coolant systems that is available for release to the containment is based on the maximum expected power operation of the core and the expected values for fuel enrichment and fuel burnup. The maximum uprated core power of 3659 MWt was used. This exceeds the 3587 MWt requested for approval by the licensee in their letter dated March 17, 2004 [Reference 1]. Therefore, the results of the dose consequence analysis that the NRC staff reviewed and approved with the AST application bound the case for this 5.2% SPU.

3.6 Materials and Chemical Engineering

3.6.1 Regulatory Evaluation

The NRC staff review in the area of materials and chemical engineering covers RV integrity, SG tube integrity, and erosion/corrosion programs. The NRC staff's review in this area focuses on the impact of proposed SPU on PTS calculations, fluence evaluations, heatup and cooldown PT limit curves, low-temperature overpressure protection, upper-shelf energy, surveillance capsule withdrawal schedules, licensee programs for addressing SG tube degradation mechanisms, and erosion/corrosion. This review is conducted to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR 50.60, 10 CFR 50.61, 10 CFR 50.55a, and 10 CFR Part 50, Appendices G and H, following implementation of the proposed SPU. Additional guidance for the NRC staff's review of the topics within the materials and chemical engineering area include the guidance contained in Chapters 4, 5, and 6 of NUREG-0800.

3.6.2 RV Material Surveillance Program

3.6.2.1 Regulatory Evaluation

The RV material surveillance program provides a means for determining and monitoring the fracture toughness of the RV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RV. Part 50 of 10 CFR Appendix H, provides the staff's requirements for the design and implementation of the RV material surveillance program. The NRC staff's review primarily focused on the effects of the proposed SPU on the licensee's RV surveillance capsule withdrawal schedule. The NRC's acceptance criteria are based on (1) GDC-14, which requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, which requires that the RCPB be designed with a margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the RV beltline region; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in SRP Section 5.3.1 and other guidance provided in Matrix 1 of NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates" (December 2003).

3.6.2.2 Technical Evaluation

Regarding the SS RPV surveillance program and capsule withdrawal schedule, the licensee concluded in Section 5.1.3 of Reference [1]:

> The calculation determined that the maximum end-of-license transition temperature shift using the SPU fluences for Seabrook Station at the end of license is less than 100 EF. These RT_{NDT} values would require three capsules to be withdrawn from Seabrook Station. Therefore, the current surveillance capsule withdrawal schedule remains acceptable for the SPU.

The licensee's calculation confirmed that the maximum end-of-license transition temperature shift using SPU fluence will remain less than 100 EF. Per American Society for Testing and Materials (ASTM) E185-82, these end-of-license transition temperature shift values would require three capsules to be withdrawn from SS, while the original withdrawal schedule in W document, WCAP-10110, March 1983 called for four capsules. Since the transition temperature shift using SPU fluence is less than 100 EF, the third capsule needs to be withdrawn not less than once or greater than twice the peak EOL fluence. The licensee has already withdrawn two capsules (U and Y). The second capsule Y was withdrawn at a peak capsule fluence of $1.15X10^{19}$ n/cm^2. The third capsule V is planned to be removed at $2.30X10^{19}$ n/cm^2, at 11.1 EFPYs. The peak vessel EOL fluence using SPU is 2.2×10^{19} n/cm^2. Hence, the licensee's third capsule withdrawal plan is within the acceptable limit of not less than once or greater than twice the peak EOL fluence. Thus, there is no impact of capsule withdrawal schedules because of SPU.

3.6.2.3 Summary

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed SPU on the RV surveillance withdrawal schedule and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the schedule. The NRC staff further concludes that the RV capsule withdrawal schedule is appropriate to ensure that the material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will provide the licensee with information to ensure continued compliance with GDC-14 and GDC-31 in this respect following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to the RV material surveillance program.

3.6.3 PT Limits and Upper Shelf Energy

3.6.3.1 Regulatory Evaluation

Part 50 of 10 CFR, Appendix G, provides fracture toughness requirements for ferritic materials (low alloy steel or carbon steel) materials in the RCPB, including requirements on the upper shelf energy (USE) values used for assessing the remaining safety margins of the RV materials against ductile tearing and requirements for calculating PT limits for the plant. These PT limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including AOOs and hydrostatic tests. The staff's review of the USE assessments covered the impact of the SPU on the neutron fluence values for the RV beltline materials and the USE values for the RV materials through the end of the current licensed operating period for SS. The NRC staff's PT limits review covered the PT limits methodology and the calculations for the number of the EFPYs specified for the proposed SPU, considering neutron embrittlement effects and using linear elastic fracture mechanics.

The NRC's acceptance criteria for PT limits and USE are based on: (1) GDC-14, which requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, which requires that the RCPB be designed with a margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of NRC RS-001, Revision 0.

3.6.3.2 Technical Evaluation

Regarding the topic of the RPV PT limits, the licensee concluded in Section 5.1.3 of Reference [1] and in the supplemental letter dated, October 12, 2004, that:

> This review indicates that the revised adjusted reference temperature (ART) after the stretch power uprate will be less restrictive than that used in developing the current ART values for Seabrook Station at 20 EFPY[s]. Therefore, no change in applicability dates is required after the stretch power uprate.

The SS TSs contain 20 EFPYs PT limit curves (refer to supplemental information provided in the licensee's letter dated October 12, 2004). These curves used a fluence of $1.324X10^{19}$ n/cm^2 (E>1.0MeV). The SPU projected fluence for 20 EFPYs is $1.12X10^{19}$ n/cm^2 (E>1.0MeV). Thus the ART values calculated for the PT limit for SPU condition are less than those used in the generation of current PT limits (20 EFPYs). Thus, there will be no impact on the PT limit curves. Therefore, the staff concludes that the licensee's proposal to limit the existing heatup and cooldown curves to a period of applicability through 20 EFPYs of operation is acceptable and consistent with the requirements of Appendix G to 10 CFR Part 50.

Regarding the topic of the RPV USE, the licensee concluded in Section 5.1.3 of Reference [1] that:

> The beltline materials were determined using the stretch power uprate fluence to have USE greater than 50 ft-lb. through the end of license, as required by 10 CFR 50, Appendix G. Based on this evaluation, the USE values for Seabrook Station will maintain a level above the 50 ft-lb. screening criterion through the end of license.

The NRC staff has evaluated the information provided by the licensee [Reference 5] as well as information contained in the staff's Reactor Vessel Integrity Database. Based on the revised SPU fluence, the staff independently confirmed that the SS RPV materials would continue to meet the USE criteria requirements of 10 CFR 50, Appendix G.

3.6.3.3 Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed SPU on the PT limits for the plant and USE values for the RV beltline materials. The staff concludes that the licensee has adequately addressed changes in neutron fluence and their impacts on the PT limits for the plant and USE values for the SS RV. The staff concludes that the SS RV beltline materials will continue to have acceptable USE, as mandated by 10 CFR Part 50, Appendix G, through the expiration of the current operation license for the facility. The NRC staff also concludes that the licensee has demonstrated the validity of the current PT limits for operation under the proposed SPU conditions. Based on this assessment, the NRC staff concludes that the SS facility will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with GDC-14 and GDC-31 in this respect following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to the PT limits and USE.

3.6.4 Pressurized Thermal Shock (PTS)

3.6.4.1 Regulatory Evaluation

The PTS evaluation provides a means for assessing the susceptibility of the RV beltline materials to PTS events to assure that adequate fracture toughness is provided for supporting reactor operation. The staff's requirements, methods of evaluation, and safety criteria for PTS assessments are given in 10 CFR 50.61. The NRC staff's review covered the PTS methodology and the calculations for the reference temperature, RT_{PTS}, at the expiration of the license, considering neutron embrittlement effects. The NRC's acceptance criteria for PTS is based on: (1) GDC-14, which requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; (2) GDC-31, which requires that the RCPB be designed with a margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (3) 10 CFR 50.61, which sets fracture toughness criteria for protection against PTS events. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of NRC RS-001, Revision 0.

3.6.4.2 Technical Evaluation

Regarding the topic of PTS analyses for the SS RPV, the licensee provided RT_{PTS} values for the beltline materials of the SS vessel [Reference 5] and concluded in their March 17, 2004, submittal:

> The pressurized thermal shock calculations were performed for the Seabrook Station beltline materials using the 10 CFR 50.61. Based on this evaluation, the pressurized thermal shock values will remain below the NRC screening criteria using the projected stretch power uprate fluence values through end of license for Seabrook Station and thus meet the requirements of 10 CFR 50.61.

The NRC staff has evaluated the information provided by the licensee as well as information contained in the staff's Reactor Vessel Integrity Database. Based on the revised SPU fluence, the staff independently confirmed that the SS RPV materials would continue to meet the PTS screening criteria requirements of 10 CFR 50.61.

3.6.4.3 Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed SPU on the PTS for the plant and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on PTS. The NRC staff further concludes that the licensee has demonstrated that the plant will continue to meet the requirements of GDC-14, GDC-31, and 10 CFR 50.61 following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to PTS.

3.6.5 Reactor Internal and Core Support Materials

3.6.5.1 Regulatory Evaluation

The reactor internals and core supports include SSCs that perform safety functions whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the RCS). The NRC staff's review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination (NDE) procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internal and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Specific review criteria are contained in SRP Section 4.5.2 and other review criteria and guidance are provided in Matrix 1 of NRC RS-001, Revision 0. Matrix 1 of NRC RS-001, Revision 0, provides references to the NRC's approval of the recommended guidelines for RV internals in Topical Reports WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals" (March 2001), and BAW-2248A, "Demonstration of the

Management of Aging Effects for the Reactor Vessel Internals" (March 2000).

3.6.5.2 Technical Evaluation

The licensee discussed the impact of the SPU on the structural integrity of the SS RV internal components in Section 5.2 of the SPU analysis report. In its safety analysis report for the SPU, the licensee concluded that the Cycle 12 SPU would not impact the safety margins associated with the structural integrity of the SS RV internal components because the SPU does not significantly increase the operating temperature for the reactor coolant (based on hot leg temperature) and SPU actually results in decrease of neutron exposures.

The RV internals of PWR-designed light-water reactors may be susceptible to the following aging effects:

- cracking induced by thermal cycling (fatigue-induced cracking), stress corrosion cracking (SCC), or irradiation assisted stress corrosion cracking (IASCC)
- loss of fracture toughness properties induced by irradiation exposure for all stainless steel grades, or the synergistic effects of irradiation exposure and thermal aging for cast austenitic stainless steel (CASS) grades;
- stress relaxation in bolted, fastened, keyed or pinned RV internal components induced by irradiation exposure and/or exposure to elevated temperatures
- void swelling (induced by irradiation exposure).

Table Matrix-1 of NRC RS-001, Revision 0, provides the staff's basis for evaluating the potential for extended power uprates to induce these aging effects. In Table Matrix-1, the staff states that, in addition to the SRP, guidance on the neutron irradiation-related threshold levels inducing IASCC in RV internal components are given in WCAP-14577, Revision 1-A. WCAP-14577, Revision 1-A, establishes a threshold of 1 X 10^{21} n/cm^2 (E $ 0.1 MeV) for the initiation of IASCC, loss of fracture toughness, and/or void swelling in PWR RV internal components made from stainless steel (including CASS) or Alloy 600/82/182 materials.

In the NRC staff's RAI #13, the staff informed the licensee that, consistent with Table Matrix-1 of NRC RS-001, Revision 0, either an inspection plan would need to be established to manage the age-related degradation in the SS RV internals, or the licensee should commit to participate in the industry's initiatives on age-related degradation of PWR RV internal components. In its response [Reference 5] to RAI #13, the licensee committed to evaluate the results of the following Electric Power Research Institute (EPRI) modification/rework package programs and to factor them into RVIs inspections as appropriate,

- Material testing of baffle/former bolts removed from the Point Beach, Farley, and Ginna nuclear power plants and determination of bolt operating parameters.
- Evaluation of the effects of irradiation, which include IASCC, swelling, and stress relaxation in PWRs.

- Evaluation of irradiated material properties.
- Void swelling assessment including available data and effects on RVIs.
- Development of a long-term RVIs aging management strategy.

The licensee’s commitment to participate in the industry’s research program of degradation of PWR RV internal components and to develop an inspection program for the RV internals that is based on the recommendations of the industry initiatives are consistent with Table Matrix-1 of NRC RS-001, Revision 0 and are, therefore, acceptable. Based on this assessment, the staff concludes that FPLE has established an acceptable course of action for managing age-related degradation in the SS RV internals under the SPU conditions for the unit.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above commitment are provided by the licensee’s administrative processes, including its commitment management program. Should the licensee choose to incorporate a regulatory commitment into the UFSAR or other document with established regulatory controls, the associated regulations would define the appropriate change-control and reporting requirements. The staff has determined that the commitment does not warrant the creation of regulatory requirements which would require prior NRC approval of subsequent changes. The NRC staff has agreed that NEI 99-04, Revision 0, “Guidelines for Managing NRC Commitment Changes,” provides reasonable guidance for the control of regulatory commitments made to the NRC staff. (See Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff," dated September 21, 2000.) The commitment should be controlled in accordance with the industry guidance or comparable criteria employed by a specific licensee. The staff may choose to verify the implementation and maintenance of this commitment in a future inspection or audit.

3.6.5.3 Summary

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed SPU on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. The NRC staff further concludes that the licensee has demonstrated that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of GDC-1 and 10 CFR 50.55a following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to reactor internal and core support materials.

3.6.5.4 Conclusion

The staff has reviewed the licensee’s proposed LAR to increase the rated core thermal power for SS by 5.2% and has evaluated the impact that the SPU conditions will have on the structural integrity assessments for the RV and RV internals. The staff has determined that the changes identified in the proposed LAR will not significantly impact the remaining safety margins required for following RCS-related structural integrity assessments: (1) RV surveillance program for SS; (2) USE assessment for the RV; (3) PT limits for the SS RV; (4) PTS assessment for the SS RV beltline materials; and (5) structural integrity assessment of the SS RV internal components, in that the licensee has committed to the establishment of a plant-specific inspection program for the RV internals.

Therefore, the NRC staff determined that the proposed power uprate will not significantly impact the operation of the RPV or the RPV internals and, therefore, the staff finds the requested power uprate acceptable with respect to reactor internal and core support materials.

3.6.6 Assessment of Material and Chemical Engineering Impacts

3.6.6.1 CVCS

3.6.6.1.1 Regulatory Evaluation

Section 9.3.4 of the licensee's UFSAR describes the CVCS. The CVCS consists of several subsystems: the charging, letdown and seal water system; the reactor coolant purification and chemistry control system, the reactor makeup control system; and the boron thermal regeneration system. The primary function of the CVCS is to maintain RCS water inventory, boron concentration, and water chemistry. In addition, the CVCS provides for RCS purification and seal injection flow to the reactor coolant pumps. During normal operation, the CVCS services the RCS by a letdown and charging process. Flow from the RCS is letdown to the CVCS then delivered back to the RCS via charging pumps. The NRC acceptance criteria for the CVCS are based on GDC-14 and GDC-29.

3.6.6.1.2 Technical Evaluation

The licensee evaluated potential impacts to the CVCS system as a result of increased core power and changes to the RCS full-load design temperatures associated with the SPU. In particular, the licensee evaluated the CVCS heat exchangers affected by potentially higher RCS operating temperatures, including the regenerative heat exchanger, letdown heat exchanger, excess letdown heat exchanger, and seal water heat exchanger. The licensee concluded that the heat duty on the CVCS heat exchangers is acceptable since the SPU operating conditions are bounded by the existing NSSS design values.

The letdown line decay time calculation requires that, during maximum flow conditions, the letdown line contains sufficient volume to delay the flow from the RCS connection to the point where it leaves the containment. The design-basis requirement for letdown line decay time is 60 seconds. The licensee stated the letdown line decay time calculated for SS is 66 seconds and this 10% margin above the required delay time is sufficient to counteract any additional N-16 activity due to the SPU. The staff agreed.

On the basis of its review, the staff concludes the CVCS heat exchangers and letdown line will continue to meet the requirements of GDC-14 and GDC-29 following the SPU since these components will continue to operate within their design limits.

3.6.6.2 SG Hardware Evaluation

3.6.6.2.1 Regulatory Evaluation

Structural evaluations were performed to determine the impact of the SPU conditions on various SG repair hardware either installed or licensed to be installed in the SS SGs. Hardware evaluations included tube mechanical plugs, weld plugs (shop and field installed), tube end machining, tube stabilizers, cut tube remnants, and significant dispositioned non-conforming conditions (i.e., loose parts). For tube plugs, the bounding applicable transient stresses (largest primary-to-secondary differential pressure) associated with plant uprate conditions and cumulative fatigue were evaluated to the ASME Code, Section III design criteria.

3.6.6.2.2 Technical Evaluation

Results from the transient stress evaluation concluded that the mechanical plug design satisfies all applicable stress and retention criteria for the power uprate condition. The mechanical plug also meets the Class 1 fatigue exemption requirements contained in Section III of the ASME Code. Stress and fatigue usage calculations for welded plugs at uprated power operation were also shown to meet the allowable ASME Code values for a 40-year design life.

Some modifications to SG tubes (e.g., removal of a tube plug by drilling/reaming prior to sleeve installation) can result in removal of a portion of the tube by a field machining process. The inside diameter of the tube may be machined to a reduction of 0.008 inches of wall thickness. Therefore, an analysis was performed to evaluate the acceptability of a tube with 0.008 inch undercut operating at the SS uprated power condition. Analysis of this maximum tube undercut condition demonstrated all stresses and fatigue usage values were within ASME Code, Section III AVs.

The licensee evaluated the straight-leg collar-cable tube stabilizer, installed in some of the SS SG tubes, for operation at SPU conditions using plant-specific geometric parameters and relative wear coefficients between the stabilizer collar and tube materials. This evaluation assumed that a random wear couple would form between the severed host tube and the stabilizer collar. Under these potentially unstable dynamic conditions, where the tube and collar start to wear, the tube wall is shown to wear through before the collar wears through. Therefore the stabilizer’s central co-axial cable remains intact for the life of the installation. Potentially deleterious contact with adjacent active tubes were found not to occur since the worn stabilizer remnant prevents significant contact with adjacent tubes.

The licensee analyzed several tube remnants in SG D to determine if operation at the power uprate conditions could result in additional tube vibration and tube remnant wear at the support plate locations. These remnants were created when some straight-leg sections of tubing were removed for metallurgical analysis in 2002. Significant tube wear at these locations could result in a cantilevered tube section contacting an adjacent active tube resulting in a primary-to secondary system leak. Based on analysis at the most limiting SPU conditions, the licensee stated the tubes will remain stable with respect to fluid elastic excitation and that turbulence induced stresses are sufficiently low to preclude crack propagation. The licensee concluded that tube stabilizers are not required for the tube remnants since these tubes have been shown to remain intact and will not contact adjacent tubes.

The licensee indicated in Reference [1] that the SS SGs contained a loose part in SG B and SG C. In response to an NRC RAI, the licensee indicated the loose part in SG B was removed during the ninth refueling outage. The loose part in SG C continues to be lodged between the same two tubes since initial observation during the first refueling outage. A loose part evaluation was performed that allows continued operation of the SGs under the uprated power conditions.

On the basis of its review, the NRC staff finds the licensee's SG hardware evaluation to be acceptable because mechanical and welded tube plugs, along with a possible tube undercut condition, were determined to meet Section III ASME Code stress limits. Also, evaluation of tube stabilizers and tube remnants without stabilizers showed they would not affect adjacent active tubes. Routine inspections performed by the licensee would detect the presence of loose parts and loose parts are evaluated on a case-by-case basis to ensure that continued operation with any foreign object is acceptable.

3.6.6.3 <u>Tube Vibration and Wear</u>

3.6.6.3.1 <u>Regulatory Evaluation</u>

An analysis was performed to evaluate the potential for increased tube wear resulting from flow induced vibration associated with operation of the SGs in an uprated power condition. Results from the current design basis vibration and wear analysis were modified to account for anticipated changes in secondary side thermal-hydraulic operating characteristics resulting from the SPU. In addition, the licensee evaluated tube stress and tube fatigue related to flow-induced vibration using the ASME Code (Section III) requirements. The NRC acceptance criteria for tube vibration and wear are based on Section III of the ASME Code and RG 1.121.

3.6.6.3.2 <u>Technical Evaluation</u>

The licensee determined that the projected increase in tube wear that may occur for the SG tubes increased from approximately 0.003 inches to approximately 0.005 inches at the uprated condition, based on a 40-year plant life. In addition, tube wear at the anti-vibration bars (AVBs) was evaluated using the most bounding SPU operating conditions. Assuming conservative wear-growth values, the licensee concluded that increased AVB wear rates due to SPU operating conditions were acceptable. Any increase in wear would progress over many cycles and would be readily observed during routine eddy current inspections.

In addition to tube wear, the licensee also evaluated tube susceptibility to fatigue at SPU conditions. Calculations performed by the licensee indicated maximum flow induced stress levels at the SPU power level resulted in negligible fatigue usage factors. SS SGs are not susceptible to the high cycle fatigue mechanism experienced by some units in the U-bend region. The SS upper-support plates are manufactured from stainless steel which will prevent the tube denting that was observed to be a necessary precursor to high-ycle fatigue.

The NRC staff finds the licensee's evaluation to be acceptable for the following reasons: the projected increase in wear is minor; any projected increase in wear will be detected during routine inspection and will be remediated to maintain tube integrity; fatigue usage factors from flow-induced vibration are negligible; and high-cycle fatigue in the U-bend section of tubing is not an issue since the necessary condition for this mechanism is precluded by use of stainless steel tube support plates.

3.6.6.4 Tube Repair Limits

NRC RG 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” describes an acceptable method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in-service inspection shall be removed from service. The allowable tube repair limit, in accordance with RG 1.121, is obtained by incorporating into the resulting structural limit a growth allowance for continued operation and an allowance for eddy current measurement uncertainty.

The structural limits for the SS SGs are provided in a W topical report, WCAP-16223, and were calculated assuming a uniform thinning mode of degradation in both the axial and circumferential directions. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in service. Analysis was performed to document applicable tube structural limits for the uprated conditions at various locations (e.g., straight-leg (free-span), support plate intersections, AVB intersections). The structural limit analysis results demonstrated sufficient margin above the tube repair limits to account for NDE uncertainty and indication growth. Therefore, the repair limit contained in the TS is adequate. The NRC staff finds the licensee’s evaluation to be acceptable because it follows the guidance in RG 1.121.

3.6.6.5 Tube Integrity Considerations

The potential effects of the 5.2% power uprate on SG tube degradation (e.g., axial and/or circumferential SCC, intergranular attack, etc.) were evaluated. Assuming reduced RCS flow and 10% of the SG tubes plugged, the maximum design reactor outlet temperature at power uprate conditions is approximately 3 EF higher than the current reactor outlet design temperature. No significant changes are expected to the primary or secondary side chemistry due to operation at SPU conditions. Although higher operating temperatures increase the propensity for SCC, the licensee concluded that the power uprate is not expected to have a significant impact on tube degradation since the licensee’s analysis projects a low amount of degradation at the end of the current license. Improved degradation resistance in the SS SGs results from: Alloy 600 thermally treated SG tubing, hydraulic expansion of the tubes in the tubesheet, stainless steel support plates with a quatrefoil-shaped hole design, and stress relief of the U-bend in rows 1 through 10. Based on the operating experience of similar design SGs, the Alloy 600 TT SG tubing has been shown to be more resistant to degradation than the Alloy 600 mill annealed SG tubing.

On the basis of its review, the staff finds the licensee’s evaluation to be acceptable since ongoing tube inspections will monitor for tube degradation and any increase in tube degradation will be addressed to ensure the regulatory requirements for tube integrity will continue to be met.

3.6.6.6 SGBS

The SGBS is designed to limit particulates and dissolved solids introduced into the SGs from the FW system. The SGBS also provides samples of the secondary side water in the SG. Proper control of SG secondary side chemistry reduces the probability of secondary side initiated SG tube degradation.

The SGBS was evaluated by the licensee to ensure it is capable of performing its intended function at SPU conditions. Since the required blowdown flow rates are based on parameters not affected by the power uprate, the blowdown flow rates required to control secondary side chemistry and SG solids will not be impacted by the SPU. The licensee stated that minor increases in FW temperatures will not negatively impact the function of the blowdown heat exchangers to cool blowdown flash tank drains below demineralizer resin temperature limits. In addition, steam generator blowdown piping will continue to be monitored for flow-accelerated corrosion (FAC). Therefore, the licensee concluded the SGBS is acceptable for operation at uprate power conditions

3.6.6.6.1 Summary

The staff has reviewed the licensee's evaluation of the effects of the power uprate on the SGBS. The staff concludes the SGBS is acceptable since blowdown flow is unchanged, SG secondary water chemistry is unchanged, and blowdown pressures and temperatures remain within the original system design.

3.6.6.7 FAC Program

FAC is a degradation mechanism caused by either single-phase or two-phase fluid flow in piping components. FAC causes wall thinning of carbon steel piping components in the power conversion system that can result in failure. Since piping system component failure may result in undesirable challenges to the plant's safety systems, the licensee maintains a program for predicting, inspecting, and repairing or replacing the components whose wall thinning could result in an FAC-related failure. In the submittal, the licensee indicated SS has not experienced excessive FAC in single-phase fluid systems. The licensee also stated that SS has experienced some FAC in two-phase flow systems, such as extraction steam and moisture separator drains. FAC rates are predicted using the CHECWORKs computer code, developed by EPRI, and ongoing inspection results are used to adjust the model predictions.

Since operating experience at SS identified the moisture separator drain piping as one of the most susceptible systems to FAC, the licensee modeled FAC rates in this piping at the SPU operating conditions. Although wear rates increased in these lines, the identified changes in wear rates are not expected to change inspection intervals significantly. The licensee stated the CHECWORKs model for SS will be updated as part of the SPU implementation and that the results of the upgraded code would be factored into the ongoing FAC program surveillance/pipe repair plans.

3.6.6.7.1 Summary

On the basis of its review, the staff finds the licensee's actions acceptable for operation under SPU conditions since the effect of the power uprate on FAC rates is expected to be small and will be adequately controlled by the existing FAC program.

3.6.6.8 Protective Coatings

Protective coatings (i.e., organic materials) inside containment are used to protect equipment and structures from the environment during normal operation and under accident conditions. The licensee stated that protective coatings inside the containment comply with the DBA testing requirements of ANSI N101.2. Although the containment coatings will be exposed to slightly higher radiation levels as a result of the SPU, they will not exceed the DBA testing values specified in ANSI 101.2. In addition, the licensee stated that other DBA conditions, such as peak temperature, peak pressure, and chemical environments will continue to be bounding for the power uprate conditions. Therefore, the licensee concluded that the protective coatings will continue to conform to RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," and will be acceptable following the SPU implementation.

3.6.6.8.1 Summary

Based on its review, the staff finds that the licensee's protective coatings program is adequate, because the parameters (e.g., temperature, pressure, radiation levels) associated with postulated SPU accident conditions continue to be bounded by the DBA assumptions. Therefore, operation under SPU conditions will not impact the protective coatings.

3.6.6.9 NSSS Piping

Under Section 5.5 of Reference [1], the licensee indicated that the maximum RCL piping stresses for the RCL piping and the corresponding code-allowable stress values were compared in accordance with the methods and code criteria as described in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," dated January 1983. This comparison is shown under Table 5.5-1 in Reference [1], "RCL Stress Analysis Summary - SPU Program," which provides a summary of the hot leg, crossover leg and cold leg maximum and allowable stresses for the following conditions: design stress, upset stress, faulted stress and thermal. The licensee discussed the impact on materials from SPU under Subsection 5.11.3 of Reference [1]. The licensee concluded that the change in the service conditions for the proposed SPU will not affect the service adequacy and performance of the NSSS piping materials.

The licensee indicated that W performed a leak-before-break (LBB) analysis of SS primary loop piping in 1984. The results of the analysis were documented in WCAP-10567, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Seabrook Units 1 and 2," which was approved by the NRC in an SE dated January 18, 1985. The licensee indicated that the RCL piping loads for LBB evaluation for the 5.2% power uprate were evaluated using the original methodology as in WCAP-10567, and found to be acceptable as shown under Subsections 5.5.2 and 5.12 of Reference [1].

The actual results are considered W proprietary information. The staff's review of the actual stresses as provided by the licensee indicated that the actual stresses due to the 5.2% power uprate are lower than the allowable stresses and therefore, acceptable for the operating temperatures generated during the 5.2% power uprate.

For the LBB evaluation, the licensee used the recommendations and criteria proposed in the "Leak-before-Break Evaluation Procedures," as stated under References 5.12-1 and 5.12-2, page 5-76, of Reference [1]. The primary loop piping dead weight, normal thermal expansion, safe shutdown earthquake (SSE) and pressure loads due to the 5.2% power uprate program normal operating temperature and pressure were used in the evaluation. The licensee concluded that all the LBB recommended margins were satisfied for the uprated power conditions.

The acceptance criteria under NRC SRP 3.6.3 recommend the following margins:

- Margin of 10 on leak rate
- Margin of 2.0 on flaw size
- Margin on loads of 1.0 (absolute summation)

The results of the licensee's evaluation showed the following:

- Actual margin of 10 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 1.0 gpm.
- Actual margin of 2.0 or more exists between the critical flaw and the flaw has a leak rate of 10 gpm.
- Actual margin of 1.0 on loads exists.

Based on the actual margins, the licensee indicated that the LBB acceptance criteria are satisfied for the SS NSSS piping at the 5.2% power uprate conditions. Furthermore, the temperatures generated during the uprate will not exceed the materials' capabilities to withstand service condition stresses.

3.6.6.9.1 Summary

On the basis of its review, the staff concludes that the impact to NSSS piping materials due to the 5.2% power uprate is minimal in terms of operating conditions, and that the materials will continue to meet the requirements of GDC-4 and GDC-14 following the SPU since these components will continue to operate within their design limits.

3.6.6.10 BOP Piping

The licensee stated in Section 8.5 of Reference [1] that an assessment of the BOP piping and supports (including main steam, extraction steam, condensate and FW, steam generator blowdown, circulating water, heater drains, service water, CCW, radwaste systems, auxiliary steam, boron recovery, chilled water, demineralized water, fire protection, fuel oil, instrument air, reactor make-up water, sampling system) was performed for a core power level of 3659 MWt. The licensee concluded that the piping and pipe supports remain in compliance with the ASME Code Section III, 1977 Edition of the "Piping Analysis Code" for its safety-related piping, and ANSI B31.1, "Piping Analysis Code," for its non-safety-related piping, and that the existing main steam piping remains acceptable for the power uprate conditions, which are based on the results of analysis with a higher flow rate resulting from the power uprate.

The licensee, in Reference [1], addressed piping system limits due to the power uprate in Section 8.5.1 and in Table 8.5-1, “Stress Summary at SPU Conditions,” for the MSS and FW System. Based on the information provided by the licensee, it showed that the actual stresses do not exceed the ASME Code allowable stress limit. In Section 8.4, Section 8.5.1, and Section 9.1.3, the licensee further discussed BOP piping under SPU. Service adequacy of the piping materials (mostly carbon steel and low-alloy steel) was evaluated for SPU operating conditions of pressure, temperature, fluid velocity, steam quality, chemistry, and where applicable, flashing conditions. The results showed that there are no changes in system water chemistry as a result of SPU, and that the existing system pipe materials, pipe size, and pipe wall thickness were appropriate and adequate for the SPU conditions. SPU operating system pressures and temperatures are bounded by existing system design conditions, and therefore, reasonable assurance is provided that the temperature and stress limits will not be exceeded for the SS BOP piping at the new service conditions.

3.6.6.10.1 Summary

Based on the information provided by the licensee, the staff concludes that the impact to the BOP piping materials due to the 5.2% power uprate is minimal, and that the BOP piping materials will continue to meet the requirements of GDC-14 following the SPU since these components will continue to operate within their design limits.

3.6.6.11 CRDMs

The existing RPV head penetrations are in the LOW susceptibility category of the First Revised NRC Order EA-03-009. The licensee stated that it would perform all required NDEs outlined in the subject Order. In Section 5.11.3, the licensee stated that there is no appreciable change in the primary water stress corrosion cracking (PWSCC) susceptibility of the CRDM nozzles from the SPU.

In an RAI to the licensee, the staff requested that the licensee discuss its evaluation made of the RPV CRDMs under the uprated conditions relative to PWSCC susceptibility. In its response, the licensee stated that the proposed SPU will result in a decrease in the vessel inlet temperature, and therefore, the CRDM nozzle materials will see less limiting conditions than they currently experience. Therefore, the PWSCC susceptibility will not increase due to the SPU.

3.6.6.11.1 Summary

Based on the information provided, the staff concludes that the CRDM materials will be acceptable for the service conditions generated under the licensee’s 5.2% power uprate, and that the CRDM materials will continue to meet the requirements of GDC-14 following the SPU since these components will continue to operate within their design limits.

3.6.6.12 Conclusion

Based on the staff's review of the information included in the licensee's submittal, the staff finds the licensee's proposed amendment acceptable with regard to the materials used in the systems identified above which comprise the RCPB piping, BOP piping, and CRDM housings and nozzles.

3.7 Human Factors

3.7.1 Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation is conducted to confirm that operator performance will not be adversely affected as a result of system changes required for the proposed SPU. The NRC staff's review covers licensee's plans for addressing changes to operator actions, human-system interfaces, and procedures and training required for the proposed SPU. The NRC's acceptance criteria for human factors are based on 10 CFR 50.54(i) and (m), 10 CFR 50.120, 10 CFR 50.34, 10 CFR 55.59, and GDC-19.

3.7.2 Technical Evaluation

The NRC staff reviewed the licensee's submittal dated March 17, 2004, and applicable supplements, for an LAR to permit an increase in the licensed core power level for operation of the SS from 3411 MWt to 3587 MWt, a power uprate increase of 5.2%. In a letter dated October 12, 2004 [Reference 5], the licensee provided additional information in response to questions from the NRC staff.

3.7.2.1 Changes in Emergency and Abnormal Operating Procedures (AOPs)

The licensee stated in its letter dated March 17, 2004 [Reference 1], "Operation at SPU conditions will result in changes to the plant's response during transients. These plant response changes are expected to result in minor adjustments to some Emergency Operating Procedures (EOPs) and Abnormal Operating Procedures (AOPs)." In its response dated October 12, 2004, the licensee provided lists of specific changes to the EOPs and AOPs associated with the SPU. The licensee further indicated that plant procedures will not require significant changes for the uprate. The same steps and sequences of steps will be maintained. In the October 12, 2004, letter, the licensee stated again that there will be no new systems or operating and maintenance procedures required by the SPU, other than one-time use procedures created for the SPU post-outage power ascension. The licensee indicated that these one-time procedures are to be used to control the power ascension and test the uprated plant in a safe and conservative manner. Further, the licensee stated that the simulator will be upgraded in both hardware and software to match the SPU design, and that the simulator core model and secondary plant models will be revised based on SPU design data. These revisions will be incorporated into the simulator before implementation in the plant to allow for operator familiarization training. The licensee also stated that operations personnel will be trained on the revised plant procedures before the proposed power uprate is implemented.

The staff finds that the licensee's response is satisfactory because no new procedures will be required, and the licensee has adequately identified and committed to implementing the necessary changes to EOPs/AOPs that will be affected by the uprate. Furthermore, the response indicated that operators will be trained on the changes before they are implemented.

3.7.1.2 <u>Changes to Operator Actions Sensitive to the SPU</u>

The licensee stated that there will be no new operator actions required by the SPU. The licensee described one action in the EOPs that will be affected by the SPU, which is a reduction in maximum hot-leg recirculation switchover time from 9 hours to 7 hours. A minimum time of 5.5 hours for this action has been established. There will be no changes to operator actions in the AOPs. The licensee stated [Reference 28]:

> Assurance is provided that hot leg injection emergency operating procedure actions will occur consistent with the calculated times because of two reasons: (1) there will be a margin of 1 hour (5 hours in the emergency operating procedures minus 4.0 hours calculated) for minimum time to hot leg injection, and a margin of 1.46 hours (7.46 hours calculated minus 6 hours in the emergency operating procedures) for maximum time to hot leg injection, and (2) based on discussions with Seabrook Station operations staff, the actions described in the hot leg injection emergency operating procedure can be performed in about 10 minutes.

In Section 3.2.13.1.4, the NRC staff documents the results of their review of the reduced time for switchover (with the times provided by the licensee in letter dated December 28, 2004, [Reference 28]) and considered it conditionally acceptable.

The staff finds that the licensee's response is satisfactory because the licensee has adequately described the effect of the power uprate on operator actions and the time limit estimates involved with the switch-over action are conservative relative to the time needed to successfully complete this action.

3.7.2.3 <u>Changes to Control Room Controls, Displays and Alarms</u>

In its March 17, 2004, letter, the licensee stated that:

> The SPU will have a limited impact on the operator interfaces for control room displays, controls, and alarms. The plant modification process will implement the required changes through normal quality program controls and the design control program. Control room indications have "band markings," that are controlled by the Operation's Department administrative control procedures and plant design control procedures. The band markings, which are mimicked in the Simulator, indicate normal operating limits, abnormal operating limits or Engineered Safety Feature actuation setpoints. The plant change packages for the uprate that identify control room changes will be processed in accordance with approved plant procedures. Training on TS changes affecting control room indications and alarms will be completed prior to implementation of the uprate.

The staff finds that the licensee's response is satisfactory because the licensee has adequately identified the changes that will occur to alarms, displays, and controls as a result of the power uprate, has adequately described how these changes will be accommodated, and that these changes will be completed before operating at SPU conditions. In addition, the licensee committed to providing training on the changes prior to SPU implementation.

3.7.2.4 Changes on the Safety Parameter Display System (SPDS)

In its letter dated October 12, 2004, the licensee provided information on the SPU effects on SPDS. Overall, the licensee indicated that there will be no changes to the layout, monitoring, or use of the SPDS as a result of the SPU. There will be some revisions to setpoints used. For example, the SPDS setpoint used to indicate that the SG narrow range level is “on span” may slightly increase above its current value. The operator actions in response to a narrow range level below this setpoint are not changing; however, the setpoint directing the operator to initiate action may increase. The licensee explained how the Operations Department (OD) has been integrated in the power uprate process by having an OD representative as a member of its power uprate team. The licensee’s design change process requires the OD review and sign-off of the design change packages. In addition, SPU design and scope have been presented to all operators as part of licensed operator requalification training classes.

The staff finds that the licensee’s response is satisfactory because the licensee has adequately identified the changes that will occur to the SPDS as a result of the power uprate and adequately described how the changes will be addressed, including the training involved.

3.7.2.5 Changes to the Operator Training Program and the Control Room Simulator

In its March 17, 2004, letter, the licensee stated “The Operations Department staff will be trained on the required modifications, TS changes, procedural changes, as well as the changes in plant response to transients and accident scenarios prior to implementation of the SPU. This will assure that the OD staff receives the required training for continued safe and reliable operations. SPU related training needs for other departments will be developed and carried out as appropriate.” In the letter dated October 12, 2004, the licensee committed to upgrading the hardware and software for the simulator to match the SPU design and provided several specific examples. The licensee will incorporate the changes to the simulator prior to plant implantations to allow for operator familiarization training. The licensee stated that the modifications were completed by the end of January 2005. The licensee plans on providing training on the modifications during the second phase of operator continuing training, which was to begin in mid-February 2005 for licensed and non-licensed operator training. Additionally, the licensee indicated that the controlling standard for the simulator is ANSI/ANS3.5-1998.

The staff considers that, based on the above licensee-described actions, the licensee will develop and implement a satisfactory training program, including simulator training, for the proposed SPU and make the necessary modifications to the simulator.

The staff finds the licensee’s response satisfactory because the licensee has adequately described how the changes to operator actions will be addressed by the simulator and training program, and how the simulator will accommodate the changes in accordance with the requirements of ANSI/ANS Standard 3.5.

3.7.3 Conclusion

The NRC staff has reviewed the licensee's planned actions related to the human factors area, and concludes that the licensee has adequately considered the impact of the proposed SPU on changes to operator actions, procedures, plant hardware, and associated training programs to ensure that operators' performance is not adversely affected by the proposed SPU. The NRC staff further concludes that the licensee will continue to meet the requirements of 10 CFR 50.54(i) and (m), 10 CFR 50.120, 10 CFR 50.34, 10 CFR 55.59 and GDC-19 following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed 5.2% SPU acceptable with respect to the human factors aspects of required system changes.

3.8 BOP

3.8.1 Introduction

The NRC staff review of proposed SPU requests focused on those areas that could potentially be affected by the proposed power uprate such that reactor safety and, consequently the health and safety of the public, could be affected. Table 1 in Enclosure 1 of Reference [3] includes a matrix that indicates whether or not the various areas of NRC staff review are affected by the proposed power uprate. Based on a review or the licensee's assessment of "what is" and "what is not" affected in the BOP areas of review, the licensee's determination appeared to be appropriate. BOP areas of review that are not affected by the proposed SPU and therefore are not specifically addressed by this evaluation include auxiliary FW, CCW, service water, the ultimate heat sink, internally generated missiles, HELB, EDG support systems, and flooding.

3.8.2 Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed SPU on (1) safe shutdown fire analyses and required systems, (2) spent fuel pool (SFP) cooling analyses and systems, (3) MSS, (4) safety-related cooling water systems, (5) condensate and FW, and (6) containment performance analyses and containment systems. The GDC contained in 10 CFR Part 50, Appendix A, apply to SS and, for the most part, establish the NRC requirements for evaluating proposed power uprates in the BOP areas of review. The licensee's conformance to NRC GDC is discussed in Section 3.1 of the SS UFSAR, as well as in the sections of the UFSAR that describe the specific SSCs that are being reviewed. In addition to compliance with the GDC, the SS UFSAR also documents licensing-basis commitments that are relevant to the performance capability of BOP systems for assuring public health and safety and protection of the environment. Acceptability of the proposed SPU for SS is based on continued compliance with the GDC; the capability of BOP systems and components to satisfy the existing plant licensing basis as described in the SS UFSAR, including commitments that have been established; and based on proposed changes or exceptions to the plant licensing basis and commitments that are appropriately recognized and justified by the licensee and found to be acceptable by the NRC staff. To the extent that the review criteria provided by RS-001, "Review Standard for Extended Power Uprates," and by NUREG-0800 (formerly NUREG 75/087), "Standard Review Plan," are consistent with the SS licensing basis, they were used to guide the staff's review efforts.

3.8.3 Safe Shutdown Fire Analyses and Required Systems

3.8.3.1 Regulatory Evaluation

The purpose of the fire protection program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that SSCs required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe shutdown following a fire. The NRC's acceptance criteria for the FPP are based on (1) 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50, insofar as they require the development of an FPP to ensure, among other things, the capability to safely shutdown the plant; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and (3) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions. Specific review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

3.8.3.2 Technical Evaluation

In RS-001, Revision 0, "Review Standard for Extended Power Uprates," Attachment 2 to Matrix 5, "Supplemental Fire Protection Review Criteria," states that

> ... power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire ... where licensees rely on less than full capability systems for fire events ..., the licensee should provide specific analyses for fire events that demonstrate that (1) fuel integrity is maintained by demonstrating that the fuel design limits are not exceeded and (2) there are no adverse consequences on the reactor pressure vessel integrity or the attached piping. Plants that rely on alternative/dedicated or backup shutdown capability for post-fire safe-shutdown should analyze the impact of the power uprate on the alternative/dedicated or backup shutdown capability ... The licensee should identify the impact of the power uprate on the plant's post-fire safe-shutdown procedures."

Section 4.1.4.3.3, "Appendix R and Safe-Shutdown Cooldown," and Section 9.1.1. "Fire Protection Program" of Attachment 1 to Reference [1] satisfactorily address these fire protection requirements of the RS-001, Revision 0. In addition, in Reference [1] it states that "systems and components are adequately sized to meet 10 CFR 50 Appendix R safe-shutdown regulatory requirements for SPU". The results of the Appendix R evaluation provided in Section 4.1.4.3.3 of Attachment 1 to Reference [1] demonstrate that the plant can be brought to a cold-shutdown condition using only safety-grade equipment following a SSE, loss of off-site power, and the most limiting single failure.

The information provided in this Section, as supplemented in response to staff RAIs, satisfactorily demonstrates the licensee's compliance. Further, the licensee indicated that the compliance with the fire protection and safe-shutdown program will not be affected because the SPU evaluation did not identify changes to design or operating conditions that will adversely impact the post-fire safe-shutdown capability. SPU evaluation does not change the credited equipment necessary for post-fire safe-shutdown nor does it reroute essential cables or relocate essential components/equipment credited for post-fire safe-shutdown. The licensee has made no changes to the plant configuration or combustible loading as a result of modifications necessary to implement the SPU that effect the FPP.

3.8.3.3 Conclusion

The NRC staff has reviewed the licensee's fire-related safe-shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe-shutdown conditions. The NRC staff further concludes that the FPP will continue to meet the requirements of 10 CFR 50.48, Appendix R to 10 CFR Part 50, and GDCs 3 and 5 following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to fire protection.

3.8.4 MSS

3.8.4.1 Regulatory Evaluation

The NRC acceptance criteria for the MSS are based on GDC-2, 4, 5, and 34. GDC-2 is related to the safety-related portions of the system being capable of withstanding the effects of various natural phenomena. GDC-4 is related to the safety-related portions of the system being capable of withstanding the environmental conditions associated to normal operations, maintenance, testing, and postulated accidents. GDC-5 is related to the sharing of safety-significant SSCs. GDC-34 is related to the residual heat removal capability of the system.

3.8.4.2 Technical Evaluation

The MSS is described in SS UFSAR Sections 10.1, 10.3, and portions of 10.4. The MSS includes both safety- and nonsafety-related functions and was designed to transport steam from the SGs to the main turbine, main feed pump turbine, emergency feed pump turbine, FW reheaters, turbine gland sealing system, steam dump system, and to the auxiliary steam system. The MSS also controls SG pressure during startup, shutdown, and when the condenser is not available; provides over-pressure protection for the SGs; isolates the containment from the main steam supply system; and provides a means to dissipate the heat generated in the NSSS during normal plant operation and following AOOs and postulated plant transients and accidents.

The licensee evaluated the MSS piping, valves, and components to verify their capability to perform at the proposed SPU conditions. The analysis included consideration of the main steam pressure and flow rate necessary to satisfy the HP turbine inlet conditions and auxiliary steam requirements; operating pressures, temperatures, and steam flow velocities for main steam piping and components; and performance requirements during all anticipated and postulated modes of MSS operation. The licensee's assessments of the following areas were reviewed by the staff to confirm that plant safety would not be degraded by the proposed power uprate:

- <u>Main Steam Piping and Components</u>
  The MSS piping and components were analyzed for a core power level of 3659 MWt (3678 MWt NSSS power level). The MSS design pressure of 1185 psig (1200 psia) bounds the proposed SPU operating conditions of approximately 1000 psia, and the highest MSS pressure and temperature (which occur at no-load conditions) are not affected by the proposed power uprate. Flow velocities through the main steam piping from the SGs to the turbine control and stop valves will increase by approximately 5% but will remain below plant and industry design limits. Based on these considerations, the staff considers the MSS piping and components to be suitable for the proposed SPU conditions.

- <u>Emergency FW Pump Turbine Steam Supply</u>
  The pressure rating of the turbine-driven emergency FW pump turbine supply and exhaust piping bounds the proposed SPU operating conditions, and the turbine-driven emergency FW pump design flow rate exceeds the SG makeup requirements for post-SPU operation. Therefore, the steam supply to the emergency FW pump turbine will continue to be adequate following the proposed uprate.

3.8.4.3 <u>Summary</u>

Based on a review of the information that was provided, the staff finds that the licensee has adequately considered and addressed the effects of the proposed power uprate on the MSS. The staff concludes that the MSS will maintain its ability to function as assumed in the UFSAR following implementation of the proposed power uprate and, consequently, SS will continue to satisfy GDC-2, 4, 5 and 34, and reactor safety will not be degraded. Therefore, the NRC staff finds that the proposed SPU is acceptable with respect to the MSS.

3.8.5 <u>Condensate and FW Systems</u>

3.8.5.1 <u>Regulatory Evaluation</u>

The NRC acceptance criteria for the condensate and FW systems are based on GDC-4, 5, and 44. GDC-4 is related to the safety-related portions of the system being capable of withstanding the environmental conditions associated to normal operations, maintenance, testing and postulated accidents. GDC-5 is related to the sharing of safety significant SSCs. GDC-44 is related to the systems capability to transfer heat from safety-related components to the ultimate heat sink.

3.8.5.2 Technical Evaluation

The condensate and FW systems return the condensate from the turbine condenser hotwells through the regenerative feed heating cycle to the SGs while maintaining the water inventories throughout the cycle. The condensate and FW systems are described in Section 10.4.7 of the SS UFSAR.

The licensee evaluated the condensate and FW systems and associated piping, pumps, valves, and pressure-retaining components to confirm their capability to operate successfully at the proposed SPU conditions. A hydraulic flow model was used to analyze and evaluate the performance of the condensate and FW systems under the proposed power uprate conditions, considering both normal plant operation and postulated transient conditions. The evaluation considered the impact of the proposed SPU on: (1) operation of the condensate, FW and heater drain pumps, including flow capacities, discharge pressures, and net positive suction head; (2) system pressures and temperatures; (3) operation of the FW heaters; and (4) the isolation capability afforded by the FW regulating and isolation valves. Based on a review of the information that was submitted, the staff found the following areas to be of interest from a reactor safety perspective:

- FW Isolation Capability
  A hydraulic flow model of the condensate and FW systems and SPU heat balances were used to predict the SPU operating conditions. These conditions were compared to the system design specifications and on this basis, the licensee determined that the proposed SPU will not impact the capability to isolate the FW system following postulated accident conditions.

- FW Flow Indication
  Indicators in the control room monitor FW flow and are banded (green color) for a normal range between 3.5 to 4.0 million pounds mass per hour. Main FW flow to each SG will increase to approximately 4.12 million pounds mass per hour at the proposed SPU conditions. Because SPU flow will exceed the existing banded range of the FW flow indicator, the indicators will be re-banded for a normal range between 3.75 and 4.25 million pounds per hour.

3.8.5.3 Summary

Based on a review of the information that was provided, the staff finds that the licensee has adequately considered and addressed the effects of the proposed power uprate on the condensate and FW systems. The staff concludes that the condensate and FW systems will maintain their ability to function as assumed in the UFSAR following implementation of the proposed power uprate and, consequently, SS will continue to satisfy GDC-4, 5 and 44, and reactor safety will not be degraded. Therefore, the NRC staff finds that the proposed SPU is acceptable with respect to the condensate and FW systems.

3.8.6 SFP Cooling

3.8.6.1 Regulatory Evaluation

The NRC acceptance criteria for the fuel storage and handling systems are based on GDC-61. GDC-61 relates to fuel storage and handling, and radioactivity control.

3.8.6.2 Technical Evaluation

The SFP cooling system provides cooling for fuel that is stored in the SFP during refueling evolutions, normal plant operating conditions, and following postulated accident conditions. The design basis of the SFP cooling system is described in Section 9.1.3 of the SS UFSAR.

Plant operation at the proposed power uprate conditions will result in an increase in the decay heat that is generated by the spent fuel which could pose a challenge to the SFP cooling system in meeting its design-basis heat transfer criteria. For plant operation at the current licensed power level and in accordance with the SS UFSAR description, the licensee performs cycle-specific analyses prior to transferring fuel from the reactor core to the SFP to confirm that the SFP bulk water temperature will not exceed its design-basis limit of 141 °F; and administrative controls have been established to provide guidance for full-core offload completion times that are less than 118.5 hours after reactor shutdown (based on the current licensed power level) to assure that the SFP bulk water temperature will be maintained at or below 140 °F. In order to accommodate the increased decay heat that will result due to SPU operation, the licensee indicated that the previously established administrative controls will be revised to provide guidance for full-core offload completion times that are less than 134 hours after shutdown to assure that the SFP bulk water temperature will continue to be maintained at or below 140 °F following SPU implementation. The heat transfer capability of the SPF cooling system in conjunction with the continued implementation of cycle-specific analyses and procedural controls that maintain the SFP temperature at or below 140 °F during full-core offloads is consistent with the licensee's current practice and provides adequate assurance that the design basis of the SFP cooling system will continue to be satisfied following the proposed power uprate. No changes to the plant licensing basis were requested by the licensee with respect to SFP cooling.

In the unlikely event there is a complete loss of cooling, the SFP bulk water temperature will begin to rise and will eventually reach the boiling temperature. The licensee has calculated that for the normal full core offload, the time required for the pool to heat up from 140 °F to boiling at 212 °F is 3.13 hours at SPU conditions versus 3.28 hours for operation at the current licensed power level. The corresponding maximum boil-off rate, based on the maximum heat load, was calculated to be approximately 104 gpm (versus 100 gpm at current conditions). The licensee stated that there is adequate time to align and supply makeup water from a variety of sources (including seismic Category 1 makeup capability) to the SFP before boiling occurs. Makeup sources include the refueling water storage tank, demineralized water, and the CVCS. All of these sources are capable of providing makeup flows in excess of 104 gpm, and are readily available and accessible. No changes to the plant licensing basis were requested by the licensee with respect to SFP makeup.

3.8.6.3 Summary

Based on a review of the information that was provided, the staff finds that the licensee has adequately considered and addressed the effects of the proposed power uprate on the capability of the SFP cooling system to perform its design-basis cooling function, and on the capability to provide makeup following a loss of SFP cooling. The staff concludes that the SFP cooling system will maintain its ability to function and that the capability to provide SFP makeup will continue to be assured following implementation of the proposed power uprate and, consequently, SS will continue to satisfy GDC-61, and reactor safety will not be degraded. Therefore, the NRC staff finds that the proposed SPU is acceptable with respect to SFP cooling.

3.8.7 Main Turbine

3.8.7.1 Regulatory Evaluation

The NRC acceptance criteria for the condensate and FW systems are based on GDC-4. GDC-4 is related to the safety-related portions of the system being capable of withstanding the environmental conditions associated to normal operations, maintenance, testing, and postulated accidents.

3.8.7.2 Technical Evaluation

As discussed in Reference [1], Attachment 1 (Section 8.3.1), and supplemented by the licensee's response to Observation No. 9 in Enclosure 1 of the submittal dated May 26, 2004, the licensee (in conjunction with General Electric Energy Services) evaluated the impact of increased throttle flow on the SS turbine-generator and related SSCs at the proposed SPU conditions. The evaluation was based on heat balance input parameters, such as SG outlet conditions, FW heater performance, and condenser pressure, as well as transient loading considerations. The licensee concluded that the existing HP turbine steam path would need to be modified to accommodate the increased steam flow that is necessary for SPU operation. The modified steam path will be designed to achieve a full throttle capability of 3719.6 MWt which will provide additional operating flexibility and design margin to ensure high standards of reliability and output at the uprated operating conditions. The licensee also plans to replace the HP turbine stage 1 nozzle plates and HP diaphragms and buckets with new, high efficiency components. Because the planned modifications will not result in any significant change in the inertia of the turbine rotor assembly, the tendency of the turbine speed to exceed design specifications following a load rejection event should not be affected by the proposed power uprate. Consequently, the current turbine overspeed trip settings will continue to be adequate for SPU operation and the vulnerability of SSCs to turbine missiles will not be affected.

3.8.7.3 Summary

Based on a review of the information that was provided, the staff finds that the licensee has adequately considered and addressed the effects of the proposed power uprate on the main turbine. The staff concludes that the main turbine will continue to be protected from overspeed conditions and the vulnerability of SSCs to turbine missiles will not be affected by the proposed power uprate and, consequently, SS will continue to satisfy GDC-4, and reactor safety will not be degraded. Therefore, the NRC staff finds that the proposed SPU is acceptable with respect to the main turbine.

3.8.8 Conclusions

The licensee has evaluated the impact of the proposed SPU on BOP systems and components, and has demonstrated that reactor safety will not be degraded at SS by the proposed power uprate to 3587 MWt. Based on the information that was provided and the considerations that are discussed in this evaluation, the NRC staff has determined that the licensee has adequately considered and addressed the effects of the proposed power uprate on the BOP areas of review, and that SS will continue to satisfy applicable GDC requirements and its current licensing basis in these areas following SPU implementation. Therefore, with respect to the BOP areas of review, the staff considers the licensee's request to increase the licensed power level of SS to 3587 MWt to be acceptable.

3.8.9 Containment Performance Analysis

3.8.9.1 Regulatory Evaluation

3.8.9.1.1 LOCA Mass and Energy Release

Section 6.4.1.1.7 of Attachment 1 to the licensee's March 17, 2004, letter (NYN-4016) cites the GDC of 10 CFR Part 50 Appendix A for the acceptance criteria for LOCA mass and energy release without citing specific GDC. Section 6.2.1.3 of the SRP [Reference 7] cites GDC-50, which requires that the reactor containment structure be designed to accommodate the calculated pressure and temperature from a LOCA without exceeding its design leakage rate. Section 6.4.1.1.7 of Attachment 1 to the licensee's March 17, 2004, letter (NYN-4016) also cites 10 CFR Part 50 Appendix K, paragraph I.A, which specifies heat sources that must be considered in satisfying the acceptance criteria of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light water nuclear power reactors." Although Appendix K is not applicable to peak pressure and temperature containment calculations, the same sources of heat are considered.

In addition, GDC-4 of 10 CFR 50 Appendix A also applies. GDC-4 requires that structures, such as the walls of subcompartments inside containment, shall be appropriately protected from the dynamic effects associated with pipe ruptures.

3.8.9.1.2 LOCA Containment Response

Section 6.4.1.1.7 of Attachment 1 to the licensee's March 17, 2004, letter (NYN-4016) cites the GDC of 10 CFR Part 50 Appendix A for the acceptance criteria. For the containment LOCA response calculations, the following GDC apply.

GDC-16, "Containment Design," requires the containment to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment following a DBA.

GDC-38, "Containment Heat Removal," requires a system to remove heat from the reactor containment. This system must rapidly reduce containment pressure (consistent with the functioning of associated systems) and be capable of withstanding a single failure. For SS, the containment spray system and the fan coolers satisfy this requirement.

GDC-50, “Containment Design Basis,” requires the reactor containment to withstand the the calculated pressure and temperature from a LOCA without exceeding its design leakage rate.

3.8.9.1.3 MSLB

For the MSLB accident, WCAP 16212-P also lists GDC 16 and 38, as described in Section 3.8.9.1.2 above, of 10 CFR Part 50 Appendix A.

3.8.9.2 Technical Evaluation

The pressure and temperature within the containment must remain below the containment’s design pressure (52 psig) and design temperature (296 EF) [Reference 8] following a postulated LOCA and a postulated MSLB at stretch power conditions.

3.8.9.2.1 LOCA Containment Response

The LOCA containment response is divided into the short term and the long-term response.

3.8.9.2.1.1 LOCA Short-Term Response

The licensee evaluated the effect of the SPU on the short-term containment LOCA response. The short-term LOCA response analysis is also termed subcompartment analysis. A subcompartment is defined in SRP Section 6.2.1.2 as a fully or partially enclosed volume within the primary containment that houses high energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume.

The NRC has approved leak-before-break technology for SS. RCS piping determined not to catastrophically rupture according to the leak-before-break technology does not have to be considered in subcompartment analyses. Consequently, as described in Section 6 of the SS UFSAR, the current licensing basis for subcompartments includes the pressurizer compartment and the pressurizer skirt cavity. The breaks for these subcompartments are the double-ended spray line break and the double-ended surge line break, respectively. Section 6.4.1.2.2 of Attachment 1 to the licensee’s March 17, 2004, letter (NYN-04016) describes the licensee’s analysis of these two breaks. For the double-ended spray line break, a conservative analysis in which a conservatively low value for the enthalpy of the break flow and a conservatively high value of the RCS pressure were used, it was found that the current releases remain bounding. For the double-ended surge line break, a sensitivity study showed that the current analysis bounds the SPU conditions.

Since SS is approved for leak-before-break and the short-term calculations for the SPU are bounded by the current analyses, compliance with GDC-4 with respect to subcompartment analysis is maintained with the SPU.

3.8.9.2.1.2 LOCA Long-Term Response

The mass and energy discharged from the break into containment are calculated using methods previously approved by the NRC [Reference 44]. Using these methods, the licensee calculated the long-term response of the SS containment to the LOCA. The containment analyses were done using the GOTHIC 7.0p2 computer code [Reference 45]. The NRC reviewed the use of the GOTHIC code for containment LOCA analyses on another docket and approved its use subject to some limitations [Reference 46]. The licensee stated that GOTHIC was used consistent with that SE [Reference 5].

Table 6.4.1.1-1 of Attachment 1 to the licensee's March 17, 2004, letter (NYN-04016) provides the initial conditions used for the containment analyses. These values have been conservatively selected.

Section 6.4.3.1 of Attachment 1 to the licensee's March 17, 2004, letter (NYN-04016) provides the calculated peak containment pressure and temperature for the LOCA. The SPU peak containment pressure is 49.13 psig following a double-ended pump suction break with one SI train and one spray train operating. This value is less than the containment design pressure of 52 psig. It is also less than the TS value of P_a [Reference 47] which is 49.6 psig. The stretch power LOCA peak containment atmosphere temperature is 271.0 EF, which is lower than both the LOCA peak containment atmosphere temperature of the current analysis (272.7 EF) and the containment design temperature (296 EF). The SPU sump water temperature is also less than the current value (258.2 EF vs. 262.5 EF).

The LOCA containment calculations for the SPU were done using acceptable methods and result in values for important containment parameters less than design values. These calculations are therefore acceptable.

3.8.9.2.1.3 Minimum Containment Pressure Analysis

The LOCA calculations performed to satisfy the safety criteria of 10 CFR 50.46(b)(1) through (5) for SS are performed using the W best-estimate LOCA model [Reference 22].

Part 50, Appendix K.D.2 of 10 CFR, "Containment Pressure," requires that for these calculations, the "containment pressure used for evaluating the cooling effectiveness during reflood...shall not exceed a pressure calculated conservatively for this purpose."

The description of the W best-estimate LOCA model states that a conservatively low value of containment pressure is calculated using current approved containment codes [Reference 22]. Since this complies with the cited section of 10 CFR Part 50, Appendix K, the staff finds the licensee's calculation of minimum containment pressure for LOCA analysis to be acceptable.

3.8.9.2.1.4 MSLB Response

The MSLB mass and energy release calculations are described in Section 6.4.4 of Attachment 1 of the license's March 17, 2004 letter (NYN-4016). The containment response is described in Section 6.4.5. The highest peak containment pressure for an MSLB under stretch power conditions (37.3 psig) occurs for a double-ended rupture at near zero power. This is less than the containment design pressure (52 psig). The peak containment temperature, 357.4 EF, is a result of a double-ended rupture of a main steamline from 100% power with failure of a diesel generator. This is less than the current peak containment temperature of 364 EF and the equipment qualification envelope temperature of 375 EF.

The MSLB calculations for SS were done using the RETRAN computer code for the mass and energy release and the GOTHIC computer code [Reference 45] for the containment calculations. The use of the RETRAN code for the analysis of the MSLB mass and energy release has been found to be acceptable [Reference 48]. GOTHIC 7.0p2 was used in a manner consistent with a previous acceptable application [Reference 46].

The licensee used conservative input values for the mass and energy calculations as described in Section 6.4.4. These are also listed in Table 6.4.4.1-2.

Since the containment response to the postulated MSLB accident was calculated with acceptable methods and conservative assumptions, and the results are within the acceptance criteria, the analyses are acceptable.

3.8.9.2.3 Post-LOCA Hydrogen Generation

Section 6.1.6 of Attachment 1 to the licensee's March 17, 2004, letter (NYN-4017) discusses the response of the SS Combustible Gas Control System to post-accident hydrogen generation at SPU conditions. The licensee concludes that, using the acceptable RG 1.7 model [Reference 49] that the resultant post-LOCA hydrogen concentrations at stretch power conditions will remain less than the 4.0 volume percent limit and are, therefore, acceptable.

3.8.9.3 Conclusion

The staff has reviewed the containment DBA analyses and finds the licensee's analysis methods and results acceptable since acceptable methods and conservative assumptions were employed and the applicable regulations remain satisfied.

3.9 Radiological Assessments

3.9.1 Regulatory Evaluation

The regulatory requirements and guidance used by the staff to evaluate the LAR are 10 CFR 20.1101 (radiation protection programs), 10 CFR 20.1301 (dose limits for individual members of the public), 10 CFR 20.1302 (compliance with the dose limits for individual members of the public), and Appendix I to 10 CFR Part 50, (numerical guides for design objectives and limiting conditions for operation to meet the criterion as low as is reasonably achievable (ALARA) for radioactive material in light-water-cooled nuclear power reactor effluents).

3.9.2 Technical Evaluation

3.9.3 Normal Operation Annual Radioactive Liquid and Gaseous Effluents

The staff has reviewed the licensee's plan for an SPU with respect to its effect on the radiological liquid and gaseous radiological effluent releases and the resultant dose to members of the public.

The licensee's liquid and gaseous radioactive waste systems are designed to maintain normal operation offsite radiological effluent releases and doses within the requirements of 10 CFR Part 20 and ALARA in accordance with Appendix I to 10 CFR Part 50.

The SPU will not change the licensee's existing radioactive waste systems (liquid or gaseous) design and plant operating procedures. Consequently, a comparison of effluent releases can be made based on current operating conditions compared to those projected for SPU inventories/radioactivity concentrations in the reactor coolant and secondary coolant and steam. The licensee used scaling techniques to assess the resultant doses to members of the public from the radiological effluent releases for the SPU.

The SPU will increase the activity level of radioactive isotopes in the primary and secondary coolant. Due to leakage or process operations, fractions of these fluids are transported to the liquid and gaseous radioactive waste systems where they are processed prior to discharge. As the activity levels in the primary and secondary coolant are increased, the activity level of inputs into the radioactive waste system are proportionately increased. The licensee has evaluated the existing radioactive waste management systems to handle the anticipated increase in activity level and concludes that they are adequate to maintain radioactive effluents ALARA.

The licensee provided evidence of their ability to maintain radioactive effluents ALARA as documented in the SS Annual Radioactive Effluent Release Reports for the past five years (1998 - 2002). These reports demonstrate that the current liquid and gaseous radioactive effluents are well within the ALARA dose criteria in Appendix I to 10 CFR Part 50.

The licensee calculated the SPU-projected increase in doses to members of the public based on operating history. The calculations show that the maximum estimated dose will be significantly below the ALARA criteria in Appendix I to 10 CFR Part 50 and the public dose limit in 10 CFR Part 20. The estimated maximum dose will also be below the calculated doses that were part of the original license in the SS Final Environmental Statement.

3.9.4 Occupation Worker Exposures

The licensee evaluated the potential increase in radiation levels in the SS as a result of the SPU. The licensee concluded that the increase in radiation levels will not significantly affect radiation zoning or shielding requirements in the various areas of the plant because it is offset by: conservative analytical techniques used to establish the original shielding requirements/design, conservatism in the current design basis reactor coolant source terms used to establish the radiation zones, and SS TSs that limit the reactor coolant concentrations to levels well below the design basis source terms.

The licensee will maintain worker radiation exposures within regulatory limits through the use of the plant's existing ALARA program, which controls access to radiation areas. Following implementation of the SPU, normal operation dose rates and available shielding will enable the licensee to continue to meet the requirements of 10 CFR Part 20 related to worker exposures and access control, and occupational radiation exposures will be maintained ALARA.

3.9.5 Conclusion

The staff has concluded that there is reasonable assurance that any increase in doses to members of the public and occupational workers after implementation of the SPU will be within the regulatory requirements in 10 CFR Part 20 and, therefore, acceptable. In addition, radioactive effluents are expected to be ALARA in accordance with Appendix I to 10 CFR Part 50.

4.0 BASES CHANGES

The licensee has provided TS Bases pages to the NRC for information. The TS Bases are not part of the TS as defined by 10 CFR 50.36. Changes to the TS Bases may voluntarily be made by a licensee in accordance with the provisions of 10 CFR 50.59. The staff did not perform an evaluation of the TS Bases revisions submitted with this application. NRC staff concurrence with the revisions is not implied by this SE. The staff may review the evaluations that support these TS Bases revisions during an inspection of SS's implementation of 10 CFR 50.59.

5.0 REGULATORY COMMITMENTS

To support the proposed SS SPU, the licensee made the following commitments, as stated:

Commitment

FPL Energy Seabrook commits to evaluate the results of the following EPRI MRP [modification/rework package] programs and to factor them into reactor vessel internals inspections as appropriate

(1)
- Material testing of baffle/former bolts removed from the Point Beach, Farley, and Ginna nuclear power plants and determination of bolt operating parameters.
- Evaluation of the effects of irradiation, which include IASCC, swelling, and stress relaxation in pressurized water reactors.
- Evaluation of irradiated material properties.
- Void swelling assessment including available data and effects on reactor vessel internals.
- Development of a long-term reactor vessel internals aging management strategy.

(2) Prior to startup from refueling outage 11, FPL Energy Seabrook commits to either upgrade the controls for the pressurizer power operated relief valves to safety-grade status and confirm the safety-grade status and water-qualified capability of the pressurizer power operated relief valves, pressurizer power operated relief valve block valves and associated piping or to provide a reanalysis of the inadvertent safety injection event, using NRC-approved methodology, that concludes that the pressurizer does not become water-solid within the minimum allowable and verifiable time for operators to terminate the event.

The NRC staff considered the above commitments as part of its evaluation in Section 3.0 and finds the commitments appropriate for the proposed SPU. Where appropriate, the NRC staff has conditioned the implementation of the proposed SPU on completion of the above commitments. Item (2) above, will be added as license condition 2.K, "Inadvertent Actuation of the Emergency Core Cooling System (ECCS)."

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Hampshire and Massachusetts officials were notified of the proposed issuance of the amendment. In a telephone call on October 26, 2004, the State officials did not have any comments.

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR Sections 51.21, 51.32 and 51.35, an Environmental Assessment and Finding of No Significant Impact was prepared and published in the *Federal Register* on February 14, 2005 (70 FR 7525). Accordingly, based upon the Environmental Assessment, the staff has determined that issuance of the amendment will not have a significant effect on the quality of the human environment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

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[3] FPL Energy Seabrook, LLC letter NYN-04016, "Response to Request for Additional Information Regarding LAR 04-03, Application for Stretch Power Uprate," dated May 26, 2004

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[5] FPL Energy Seabrook, LLC letter SBK-L-04072,"Response to Request for Additional Information Regarding LAR 04-03, Application for Stretch Power Uprate," dated October 12, 2004

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[22] WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), “Code Qualification Document for Best Estimate LOCA Analysis,” March 1998

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Attachment: List of Acronyms

LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
Aac	alternate alternating current
ac	alternating current
ADAMS	Agencywide Documents Access and Management System
AL	analytical limit
ALARA	as low as is reasonably achievable
AMSAC	anticipated transient without scram mitigating system actuation circuitry
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AOP	abnormal operating procedures
AOV	air-operated valve
ART	adjusted reference temperature
ASME Code	American Society of Mechanical Engineers Boiler and Pressure Vessel Code
AST	alternate source term
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AV	allowable value
AVB	anti-vibration bar
BOL	beginning of life
BOP	balance-of-plant
BTP	Branch Technical Position
CASS	cast austenitic stainless steel
CCW	component cooling water
CRDM	control rod drive mechanism
CRDS	control rod drive system

CSS	core support structure
CST	condensate storage tank
CUF	cumulative usage factor
CVCS	chemical and volume control system
DBA	design-basis accident
dc	direct current
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFPY	effective full-power year
EOL	end of life
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EQ	environmental qualification
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
FAC	flow-accelerated corrosion
FPLE	FPL Seabrook, LLC (licensee)
FPP	fire protection program
FW	feedwater
FWP	feedwater pump
GDC	general design criteria
GL	generic letter
GOTHIC	Generation of Thermal-Hydraulic Information for Containment
GSU	generator step-up
H_3BO_3	boric acid
HELB	high-energy line break

HP	high pressure
I&C	instrumentation and control
IASCC	irradiation assisted stress corrosion cracking
ICSB	Instrumentation & Control Systems Branch
IPB	isolated phase bus
ISA	Instrument Society of America
ISO-NE	Independent System Operator-New England
LAR	license amendment request
LBB	leak before break
LBLOCA	large-break loss-of-coolant accident
LOAC	loss of ac
LOCA	loss-of-coolant accident
LONF	loss of normal feedwater
LOOP	loss-of-offsite power
LSSS	limiting safety system setting
LTC	long-term cooling
LTOP	low-temperature overpressure protection
MMF	minimum measured flow
MOV	motor-operated valve
MSLB	main steamline break
MSS	main steam system
MSSV	main steam safety valves
MTC	moderator temperature coefficient
MVA	megavolts-amperes
MVAR	megavolt-ampere reactive
MWe	megawatts electric
MWt	megawatts thermal
NEI	Nuclear Energy Institute

NEPOOL	New England Power Pool
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OD	Operations Department
PCT	peak cladding temperature
PORV	power-operated relief valve
PT	pressure-temperature
PTS	pressurized thermal shock
PWR	pressurized-water reactor
PWSCC	primary water stress corrosion cracking
RAI	request for additional information
RAT	reserve auxiliary transformer
RCCA	rod cluster control assembly
RCL	reactor coolant loop
RCP	reactor coolant pressure
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RPS	reactor protection system
RPV	reactor pressure vessel
RTDP	revised thermal design procedure
RTP	rated thermal power
RV	reactor vessel
RVI	reactor vessel internal
SAFDL	specified acceptable fuel design limit
SAL	safety analysis limit
SBLOCA	small-break loss-of-coolant accident

SBO	station blackout
SCC	stress corrosion cracking
SE	safety evaluation
SER	safety evaluation report
SFP	spent fuel pool
SG	steam generator
SGBS	steam generator blowdown system
SI	safety injection
SIS	safety injection system
SL	safety limit
SOV	solenoid-operated valve
SPDS	safety parameter display system
SPU	stretch power uprate
SRP	Standard Review Plan
SS	Seabrook Station
SSE	safe shutdown earthquake
SSC	structure, system, and component
STDP	standard thermal design procedure
TA	total allowance
TDF	thermal design flow
TS	technical specification
UAT	unit auxiliary transformer
UFSAR	updated final safety analysis report
USE	upper shelf energy
W	Westinghouse