January 31, 2003

Mr. Paul D. Hinnenkamp
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
P. O. Box 220
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, ISSUANCE OF AMENDMENT RE: 1.7 PERCENT INCREASE IN LICENSED POWER LEVEL (TAC NO. MB5094)

Dear Mr. Hinnenkamp:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 129 to Facility Operating License (FOL) No. NPF-47 for the River Bend Station, Unit 1. This amendment consists of changes to the Technical Specifications and FOL in response to your application dated May 14, 2002, as supplemented by letters dated July 9, August 2, September 16, and November 7 and 22, 2002.

This amendment increases the licensed power level by approximately 1.7 percent from 3,039 megawatts thermal (MWt) to 3,091 MWt. These changes result from increased feedwater flow measurement accuracy to be achieved by utilizing high-accuracy ultrasonic flow measurement instrumentation.

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

Sincerely,

Michael K. Webb, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 129 to NPF-47
2. Safety Evaluation

cc w/encls: See next page

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**NLO with comment
* SE input provided - no major changes made.

Accession No.: ML030340294

OFFICE	PDIV-1/PM	PDIV-1/LA	EMCB/SC	EMCB/SC	SPSB/SC(A)	SRXB/SC	EMEB/SC
NAME	MWebb	DJohnson	LLund*	SCoffin*	FMReinhart*	RCaruso*	KManoly*
DATE	1/23/2003	1/23/03	12/20/02	08/15/02	07/18/02	07/24/02	12/23/02

OFFICE	EEIB/SC	EEIB/SC	OGC**	PDVI-1/SC	PDVI/D	DLPM/D
NAME	CHolden*	EMarinos*	RWeisman	RGramm	WRuland	TMarsh for JZwolinski
DATE	10/24/02	10/23/02	31 Jan 03	1/30/03	1/30/03	1/31/03

OFFICIAL RECORD COPY

River Bend Station

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ENTERGY GULF STATES, INC. **

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 129
License No. NPF-47

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

   A. The application for amendment by Entergy Gulf States, Inc.* (the licensee) dated May 14, 2002, as supplemented by letters dated July 9, August 2, September 16, and November 7 and 22, 2002, 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, as amended, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

---

* Entergy Operations, Inc. is authorized to act as agent for Entergy Gulf States, Inc., and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

**Entergy Gulf States, Inc., has merged with a wholly owned subsidiary of Entergy Corporation. Entergy Gulf States, Inc., was the surviving company in the merger.

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.

2. Accordingly, the license is amended by changes to the Technical Specifications and the Facility Operating License as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-47 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 129 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by L. Marsh for/

John A. Zwolinski , Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License and
Technical Specifications

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NO. 129

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of the Facility Operating License and 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	Insert
LICENSE	
page -3-	page -3-
TECHNICAL SPECIFICATIONS	
1.0-5	1.0-5
3.4-1	3.4-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By application dated May 14, 2002 (Reference 8.1), as supplemented by letters dated July 9, August 2, September 16, and November 7 and 22, 2002, (References 8.1a through 8.e, respectively), Entergy Operations, Inc. (EOI, Entergy, or the licensee) submitted a request for changes to the River Bend Station, Unit 1 (RBS), Facility Operating License (FOL) and Technical Specifications (TSs). This proposed amendment would increase the licensed power level by approximately 1.7 percent from 3,039 megawatts thermal (MWt) to 3,091 MWt. These changes are based on increased feedwater (FW) flow measurement accuracy to be achieved by utilizing high-accuracy ultrasonic flow measurement instrumentation.

The supplemental letters dated July 9, August 2, September 16, and November 7 and 22, 2002, provided clarifying information that did not change the scope of the original *Federal Register* notice (67 FR 40022, published June 11, 2002) or the original no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

Nuclear power plants are licensed to operate at a specific thermal power level. The power level is indicated in the control room by neutron flux instrumentation that is calibrated to correspond to core thermal power. Core thermal power level is determined by a nuclear steam supply system (NSSS) energy balance calculation. The accuracy of this calculation depends primarily upon the accuracy of FW flow, temperature, and pressure measurements. The thermal power levels assumed in the plant design basis transient and accident analyses must bound the potential range of power levels at which the plant could be operated. The uncertainty of calculated values of core thermal power levels is factored into the allowable thermal power levels to reduce the likelihood of exceeding the power levels assumed in the analyses. Before June 1, 2000, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix K, “ECCS [Emergency Core Cooling System] Evaluation Models,” required licensees to base their transient and accident analyses on an assumed power level of 102 percent of licensed thermal power. This was to allow for uncertainties in determining thermal power (e.g., instrument error uncertainties).

On June 1, 2000, a revision to 10 CFR Part 50, Appendix K, was issued, to be effective on July 31, 2000. The stated objective of this rulemaking was to reduce an unnecessarily burdensome regulatory requirement. Appendix K was originally issued to ensure an adequate performance margin for the ECCS in the event of the occurrence of a design basis loss-of-coolant accident (LOCA). The margin is provided by conservative features and requirements of the evaluation models, and by the ECCS performance criteria. As stated above, the original regulation did not require the power measurement uncertainty to be demonstrated, but rather mandated a 2 percent margin. The new rule allows licensees to justify a smaller margin for power measurement uncertainty. Because there will continue to be substantial conservatism in other Appendix K requirements, a sufficient margin to ECCS performance in the event of a LOCA will be preserved.

However, the final rule, by itself, did not allow increases in licensed power levels. Because the licensed power level for a plant is a licensed limit, proposals to raise the licensed power level must be reviewed and approved under the license amendment process. References 8.1 and 8.1e include justifications of the reduced power measurement uncertainty and the basis for the modified ECCS analysis.

RBS was originally licensed to operate at a maximum power level of 2,894 MWt. Amendment No. 114, issued on October 6, 2000, authorized a power uprate to 3,039 MWt, to which a 2 percent margin is added in the ECCS evaluation model to allow for uncertainties in the core thermal power measurement, as was previously required by 10 CFR Part 50, Appendix K. For that uprate, Entergy did not request to apply the Appendix K revision permitting licensees to use an assumed power level less than 1.02 times the licensed power level, provided the new power level is demonstrated to account for uncertainties due to power level instrument error.

RBS intends to install the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ (✓+™) System for FW flow measurement. As described below, use of the LEFM✓+ ™ System will reduce the calorimetric core power measurement uncertainty to < 0.3 percent. Based on this, Entergy is proposing to reduce the power measurement uncertainty required by 10 CFR Part 50, Appendix K, to permit an increase of 1.7 percent in the licensed power level. The reduction in power measurement uncertainty does not constitute a significant change to the ECCS evaluation model as defined in 10 CFR 50.46(a)(3)(i).

Use of the LEFM ✓+ ™ System provides a more accurate measurement of FW flow than the instrumentation originally installed at RBS. Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy While Increasing Power Level Using The LEFM System," and its supplement, Engineering Report ER-157P, "Supplement to Caldon Topical ER-80P: Basis for a Power Uprate With the LEFM✓™ or LEFM CheckPlus ™ System," Revision 5, were approved by the staff in March 1999 (Reference 8.2) and December 2001 (Reference 8.3), respectively.

The plant-specific basis for the proposed uprate is provided in the applicable sections of the General Electric Company (GE) Nuclear Energy topical report included in Reference 8.1e and in References 8.1b, 8.1c, and 8.1d.

2.1 Applicable Regulatory Requirements/Criteria

The staff finds that the licensee, in Sections 3.0, 4.0, and 5.0 of Attachment 1 to Reference 8.1, identified the applicable regulatory requirements. In its review of the requested action, the staff considered the requirements of 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 17; 10 CFR Part 50, Appendix G; 10 CFR Part 50, Appendix K; 10 CFR Part 50, Appendix R;

10 CFR 50.46; 10 CFR 50.49; 10 CFR 50.63; 10 CFR 50.90; and 10 CFR 50.92 for no significant hazards consideration determinations and TSs.

3.0 TECHNICAL EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment, which are described in Sections 3.0, 4.0, and 5.0 of Attachment 1 to Reference 8.1 for power uprate. The detailed evaluation below will support the conclusions 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.

The NRC staff's review of the licensee's application is organized as follows:

3.1	Reactor - Core and Fuel Performance
3.1.1	Fuel Design and Operation
3.1.2	Thermal Limits Assessment
3.1.3	Reactivity Characteristics
3.1.4	Stability
3.1.5	Reactivity Control
3.1.5.1	Control Rod Drives and Control Rod Drive Hydraulic System
3.1.5.2	Control Rod Drive Mechanisms
3.2	Reactor Coolant System and Connected Systems
3.2.1	Nuclear System Pressure Relief/American Society of Mechanical Engineers Boiler and Pressure Vessel Code Overpressure Protection
3.2.2	Reactor Pressure Vessel and Internals
3.2.3	Reactor Vessel Fracture Toughness
3.2.4	Reactor Coolant Piping Components
3.2.4.1	Reactor Coolant Pressure Boundary Piping
3.2.4.2	Balance-of-Plant Piping and Safety-Related Valves
3.2.4.3	Flow-Accelerated Corrosion in Piping
3.2.5	Reactor Recirculation System
3.2.6	Main Steam Isolation Valves and Main Steamline Flow Restrictors
3.2.7	Reactor Core Isolation Cooling System
3.2.8	Residual Heat Removal System
3.2.9	Reactor Water Cleanup System
3.3	Engineered Safety Features
3.3.1	Containment System Performance
3.3.2	Emergency Core Cooling Systems
3.3.2.1	High-Pressure Core Spray System
3.3.2.2	Low-Pressure Core Spray System
3.3.2.3	Low-Pressure Coolant Injection Mode
3.3.2.4	Automatic Depressurization System
3.3.3	Emergency Core Cooling System Performance Evaluation
3.3.4	Main Control Room Atmospheric Control System
3.3.5	Standby Gas Treatment System and Main Steam Positive Leakage Control System

3.1 Reactor - Core and Fuel Performance

The licensee submitted licensing report NEDC-33051P, Revision 1, "Safety Analysis Report for River Bend Station, Unit 1 Thermal Power Optimization" (Attachment 1 to Reference 8.1e), to support the proposed power uprate. The report evaluated the impact of the increased operating power on the facility's safety analyses and on the capabilities and performance of the NSSS and its components. The power-dependent safety analyses, which are calculated at 102 percent of the current reactor thermal power, will remain applicable and bounding at the uprated condition; however, other analyses and equipment or system qualifications performed at nominal power have to be reevaluated. The licensee stated that the power uprate will be achieved by increasing the FW flow to produce higher steam flow from the reactor vessel and by adjusting the turbine control valve (TCV) position to reduce the main steam (MS) line flow resistance.

The licensee's power uprate submittal follows the generic format and content of the boiling water reactor (BWR) power uprate licensing topical report, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization (TLTR)," Licensing Topical Report NEDC-32938P (Reference 8.4 or TLTR). This report is under staff review and is intended to be used for reference in future plant-specific thermal power optimization (TPO) requests. Reference 8.4 is referred to in several sections of the RBS plant-specific TPO Safety Analysis Report (TSAR), even though the TLTR covers power uprates only to 1.5 percent. In response to a staff question regarding the applicability of the TLTR to the 1.7 percent power uprate, in Reference 8.1d Entergy provided a discussion of its methodology for the TPO analysis. The methodology followed one of three approaches: (a) the existing analysis was conducted at 102 percent or greater of current licensed thermal power (CLTP) and, therefore, bounded the uprate conditions; (b) a new plant-specific analysis was conducted; or (c) the licensee confirmed that the generic analysis methodology is applicable to RBS at the uprated power level. In those TSAR sections for which the TLTR did not bound the uprate, Entergy used methods (a) or (b) to perform an analysis to determine whether the system(s) discussed in that section could continue to perform their functions at the proposed uprated condition. The staff accepts Entergy's methodology for the references to the TLTR in the RBS TPO report (and the attendant section-specific evaluations).

The current Cycle 11 RBS core utilizes a mixed core of 624 fuel assemblies, which consists of 200 Framatome (formerly known as Siemens Power Corporation) ATRIUM-10 bundles and 424 GE 11 bundles. The RBS reload analysis is based on the NRC-approved GE methodology described in NEDE-24011-P-A-14, "General Electric Standard Application for Reactor Fuel (GESTAR II)" (Reference 8.5) and Framatome methodologies described in ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model" (Reference 8.6). The NRC-approved codes and methodologies used for the licensing safety analyses are also referred to in Section 5.0 of the RBS TSs. The limiting anticipated operational occurrence (AOO) and accident analyses are reanalyzed for every reload, and the safety analyses are documented in Chapter 15 of the Updated SAR (USAR). Limiting AOOs and accidents are events that could potentially affect the core operating and safety limits that ensure the safe operation of the plant.

The core thermal-hydraulic design and fuel performance characteristics are evaluated for each fuel cycle in accordance with the NRC-approved Framatome and GE design criteria, analytical models, and methods described in References 8.5 and 8.6.

The following sections address the effect of the power uprate on fuel design performance, thermal limits, the power/flow map, and reactor stability.

3.1.1 Fuel Design and Operation

Fuel bundles are designed to ensure that (1) the fuel bundles are not damaged during normal steady-state operation and AOOs, (2) any damage to the fuel bundles will not be so severe as to prevent control rod insertion when required, (3) the number of fuel rod failures during an accident is not underestimated, and (4) the coolability of the core is always maintained. For each fuel vendor, the NRC-approved fuel design acceptance criteria and analysis methodology assure that the fuel bundles comply with the objectives of Sections 4.2 and 4.3 of the Standard Review Plan (Reference 8.7) and the applicable GDC of 10 CFR Part 50, Appendix A. The fuel vendors perform thermal-mechanical, thermal-hydraulic, neutronic, and material analyses to ensure that the fuel system design can meet the fuel design limits during steady-state, AOO, and accident conditions.

The uprated core for RBS will consist of GE 11 and Framatome ATRIUM-10 fuel bundles. The fuel design criteria are based on the NRC-approved methodology described in ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs" (Reference 8.8), and GESTAR II. A new mechanical fuel design is not needed to achieve the 1.7 percent power uprate, though new fuel designs may be used in the future to obtain additional operating flexibility or to maintain the fuel cycle length. The current GE 11 and ATRIUM-10 fuel meets the NRC-approved acceptance criteria, and, should the licensee seek to use any new fuel designs that do not comply with the NRC-approved fuel design criteria given in References 8.5 and 8.8, the licensee will need to submit an amendment request for NRC review of such a proposal.

The slightly higher operating power and the increased steam void content will affect the core and fuel performance. The licensee may also change the power distribution in the reload design to allow more operating flexibility or to maintain the fuel cycle length. This would also affect the core and fuel performance. However, the steady-state and transient design linear heat generation rate limits for each fuel bundle ensure that the fuel plastic strain design limit or the fuel centerline melt limit will not be exceeded. The thermal-hydraulic design and the operating limits will also ensure that the probability of boiling-transition fuel failures will not increase at the uprated conditions.

When a new fuel type is introduced, numerous evaluations are performed as part of the reload process. These evaluations confirm, among other things, that maximum fuel burn up which any particular topical report approved is not exceeded. In addition, the evalutions address all other impacts this new fuel type may have on operation at the TPO power level, including impacts on stability, thermal-hydraulic compatibility, radiological analyses, and hydrogen generation. The licensee will follow acceptable methods and processes described in approved fuel vendor topical reports to perform these analyses and evaluations.

3.1.2 Thermal Limits Assessment

GDC 10, "Reactor Design," of 10 CFR Part 50, Appendix A, requires that the reactor core and the associated control and instrumentation systems be designed with appropriate margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents). The safety limit minimum critical power ratio (SLMCPR) protects 99.9 percent of the fuel rods from boiling transition during steady-state operation. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as result of an AOO. The operating linear heat generation rate (LHGR) is the core operating limit that assures the fuel thermal-mechanical performance limit (i.e., the 1 percent fuel plastic strain design limit or the no-fuel-centerline-melt criterion) will not be exceeded as a result of an AOO.

The SLMCPR is calculated for every reload at the rated thermal power (RTP) using NRC-approved methodologies. In FOL Amendment Number 122, dated October 3, 2001 (Reference 8.13), the staff approved the minimum critical power ratio (MCPR) safety limit for the current fuel Cycle 11. The staff concluded that Entergy's justification for analyzing and determining the SLMCPR value of 1.08 for two-recirculation-loop operation and 1.10 for single loop operation was acceptable for RBS Cycle 11, since approved methodologies were used.

The OLMCPR is determined on a cycle-specific basis from the results of the reload transient analysis and this approach will not change. AOOs are analyzed at various points in the allowable operating domain, depending on the type of transient. The change in the MCPR is combined with the SLMCPR to establish the OLMCPR, which ensures that 99 percent of the rods will not reach boiling transition in the event of an anticipated transient. The licensee will calculate the OLMCPR at the uprated condition for RBS in cycle-specific reload analyses, and these limits will be maintained during operation.

The steady-state and transient LHGR limits are established for every fuel design to protect against fuel centerline melt throughout the operating cycle. The licensee will determine the LHGR limits for the uprated power in the reload analysis for future cycles, and these limits will be maintained during operation.

The maximum planar LHGR (MAPLHGR) operating limit is based on the most limiting LOCA and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, the licensee performs LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload, the licensee confirms that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

Thus, the licensee will calculate the OLMCPR, the SLMCPR, the LHGR, and the MAPLGHR for the uprated conditions as part of the reload analysis required under the TSs using NRC-approved methodologies. The licensee will propose appropriate changes to the limits in the TSs and/or the core operating limit report resulting from these reload analyses. Accordingly, the staff concludes that these limits will remain appropriately controlled.

3.1.3 Reactivity Characteristics

The reload core analysis will ensure that the minimum shutdown margin requirements will be met for each core design, which the staff finds acceptable.

3.1.4 Stability

RBS utilizes reactor stability Enhanced Option-1A (EIA). The EIA absolute high flow control line (which is used in the stability region boundary validation) does not change for the TPO uprate. Therefore, there is minimal effect on stability beyond the normal cycle-to-cycle core characteristic variations that are evaluated with the reload. The TPO uprate does not significantly affect stability. Reload stability evaluations continue to ensure acceptable stability performance and protection for future cores operating at TPO uprate conditions. The staff finds this acceptable.

3.1.5 Reactivity Control

3.1.5.1 Control Rod Drives and Control Rod Drive Hydraulic System

The generic discussions in Reference 8.4, Section 5.6.3 and Section 2.3.3 of Appendix J, apply to RBS. The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The scram and rod insertion/withdrawal functions of the CRD system depend on the operating reactor pressure, and the pressure difference between the CRD system hydraulics and the reactor vessel bottom head pressure. Entergy determined that the CRD system is capable of performing its design functions of rapid rod insertion (scram) and rod positioning (insertion/withdrawal).

The staff finds that the proposed power uprate will not have a significant impact on the operation of the CRD system for the following reasons:

(1) The operating dome pressure will not change, and the scram timing at steady-state power conditions will not be affected.

(2) There must be a minimum pressure differential of 250 pounds per square inch (psi) differential between the hydraulic control unit (HCU) and the vessel bottom head for normal CRD insertions and withdrawals. Since the operating dome pressure will not increase, the power uprate will have little impact on the CRD pump capacity.

Therefore, the staff finds that the CRD system will continue to perform all its safety-related functions at the proposed uprated conditions.

3.1.5.2 Control Rod Drive Mechanisms

The licensee evaluated the CRD mechanisms (CRDMs) by comparing the proposed parameters to those in the design basis analysis. The licensee indicated that the reactor vessel operating and design pressure and temperature that are used in the existing design basis analysis remain bounding for the proposed 1.7 percent power uprate. The licensee concluded that the existing RBS design basis for stresses and fatigue cumulative usage factors (CUFs) of the CRDMs are not affected by the proposed 1.7 percent power uprate condition. For the same reasons, the staff concludes that the CRDMs will continue to meet its design basis and performance requirements for the proposed 1.7 percent power uprate condition.

3.2 Reactor Coolant System and Connected Systems

3.2.1 Nuclear System Pressure Relief/American Society of Mechanical Engineers Boiler and Pressure Vessel Code Overpressure Protection

The safety/relief valves (SRVs) provide overpressure protection for the NSSS during abnormal operational transients. The steam flow associated with the 1.7 percent power uprate can be regulated adequately by adjusting the TCV position; therefore, the operating dome pressure will not increase, and the SRV setpoints and the number of valve actuation groups will not be changed.

Table 1-2 of Attachment 1 to Reference 8.1b provides the thermal-hydraulic parameters for the rated and the proposed uprated conditions. The tables show that for a core flow of 107 percent, the steam flow rate increases by 1.7 percent for the uprated conditions.

The RBS Cycle 11 overpressure protection analysis was performed with the NRC-approved COTRANSA2 methodology in AN-913 (P) (A) (Reference 8.10).

Since the SRVs will actuate at the current setpoints and the current American Society of Mechanical Engineers (ASME) overpressure protection analysis is based on operation at 102 percent power, the staff accepts the licensee's assessment that the SRVs will have sufficient capacity to handle the increased steam flow associated with the proposed uprate.

3.2.2 Reactor Pressure Vessel and Internals

The licensee evaluated the reactor vessel and internal components, considering the changes of the design input parameters and loads due to the proposed 1.7 percent power uprate. The loads applicable to the internal components include reactor internal pressure difference (RIPD), LOCA, SRV, seismic, annulus pressurization, jet reaction, and fuel lift loads.

The licensee evaluated the reactor vessel components in accordance with the ASME Boiler and Pressure Vessel Code (ASME Code), 1971 Edition with addenda to and including Summer 1973, which is the code of record. In Reference 8.1c, the licensee indicated that the current licensing basis analysis for the reactor vessel was performed at 102 percent of the CLTP or 3,100 MWt, for normal, upset, emergency, and faulty conditions. Therefore, the licensee concluded that the existing licensing basis analysis for the reactor vessel remains bounding for the proposed power uprate at 101.7 percent of the current rated power or 3,091 MWt. The staff finds the licensee's conclusion acceptable.

The licensee evaluated the reactor internals due to the slight increase in the FW flow and temperature, and the RIPDs. The calculated stresses and fatigue CUFs for the affected limiting reactor internals are shown in Table 3-2 of Attachment 1 of Reference 8.1e. The calculated stresses and CUFs provided for the 1.7 percent power uprate remain below the code-allowable limits and are, therefore, acceptable to the staff.

The licensee assessed flow-induced vibration for the proposed power uprate for limiting reactor internal components. As a result of its evaluation, the licensee concluded that vibration of all safety related internal components due to flow induced vibration loads will remain within the GE acceptable stress limits of 10 kilo-psi (ksi). The staff accepts the licensee's conclusion that the reactor internals will remain adequate and acceptable for the proposed 1.7 percent power uprate, considering the conservative GE acceptable limits of 10 ksi in comparison to the ASME allowable limit of 13.6 ksi for service cycles equal to 1×10^{11}.

Based on its review of the licensee's evaluation of the reactor vessel and internals as set forth above, the staff concludes that the reactor vessel and internal components will continue to maintain their structural integrity for the proposed 1.7 percent power uprate condition.

3.2.3 Reactor Vessel Fracture Toughness

Appendix G of 10 CFR Part 50 contains several fracture toughness requirements. The appendix states that beltline materials must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb. Table 3 of Appendix G contains the pressure-temperature (P-T) limits and minimum temperature requirements that are defined by the operating conditions of the reactor vessel.

Appendix H of 10 CFR Part 50 contains the requirements for the reactor vessel material surveillance program. The criteria in the appendix states that reactor vessels that have peak neutron fluence that exceeds 10^{17} n/cm^2 at the end of the design life must have their beltline materials monitored by a surveillance program complying with American Society for Testing and Materials (ASTM) E 185, as modified by Appendix H. The design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of ASTM E 185 that was current at the time the reactor vessel was purchased. The licensee evaluated the fracture toughness of the reactor pressure vessel (RPV) using NRC-approved fracture toughness evaluation procedures in Section 3.2.1 of Reference 8.4. The end-of-life (EOL) fluence was calculated for the TPO uprate conditions, and the fluence for current conditions was used to evaluate the vessel against the requirements of 10 CFR Part 50, Appendix G. The results of the licensee's evaluations indicate that:

(1) The upper shelf energy (USE) values for all RBS beltline materials remain greater than 50 ft-lb through the end of the facility's current operating license. The staff performed confirmatory calculations and agrees that the minimum EOL USE for beltline materials is 67 ft-lb. Based upon the technical evaluation above, the staff concludes that the USE values for all RBS beltline materials remain above 50 ft-lb through 32 effective full power years (EFPYs) at the proposed uprated power level. Hence, the RBS RPV beltline materials comply with the USE requirements in 10 CFR Part 50, Appendix G.

(2) The current 32 EFPY P-T curves remain valid with the TPO uprate. The RPV surface fluence increases due to the TPO uprate. The net effect, however, in 1/4T fluence at 32 EFPY is negligible after implementation of the TPO. Because the 1/4T fluence contributes to the resulting adjusted reference temperature (ART), there is no change to ART or Shift for EFPYs up to and including 32 EFPY. The P-T curves currently licensed for RBS for 32 EFPY account for a Shift value of 152 ^{o}F. The Shift values calculated for the TPO are unchanged up to 32 EFPY. Hence, the RBS RPV P-T limits continue to comply with the requirements of Appendix G through 32 EFPY. The staff agrees that with negligible changes to the 1/4T fluence, the current 32 EFPY P-T curves remain valid with the TPO uprate.

(3) The reactor vessel material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H after the TPO. The RBS surveillance program consists of three capsules that have been in the reactor vessel since plant startup. One of these capsules was removed after approximately 10.08 EFPY of operation, the second capsule is scheduled to be removed at 15 EFPY, and the third capsule is classified as "Standby." The licensee concluded that the TPO uprate has no effect on the existing surveillance schedule. The staff accepts the licensee's conclusion that, with negligible changes in fluence, the reactor vessel material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H, after the TPO has been implemented.

The staff concludes that the licensee's proposed amendment to implement a 1.7 percent power uprate at RBS follows the requirements set forth in Appendices G and H of 10 CFR Part 50. Based on its review of information included in References 8.1 through 8.1e, the staff finds the licensee's proposed amendment acceptable.

3.2.4 Reactor Coolant Piping and Components

3.2.4.1 Reactor Coolant Pressure Boundary Piping

The licensee evaluated the effects of the proposed 1.7 percent power uprate condition on the reactor coolant piping, components, and their supports with regard to changes in flow rate, temperature, and pressure. The licensee summarized its evaluation of reactor coolant pressure boundary (RCPB) piping inside the containment in a table in Section 3.5.1 of Attachment 1 to Reference 8.1e. The evaluated piping systems include recirculation, MS and attached piping systems (including SRV discharge line), reactor core isolation cooling (RCIC) piping, MS drain lines, RPV head vent line, FW piping (inside containment), RPV bottom head drain line, residual heat removal (RHR), low pressure core spray (LPCS), high pressure core spray (HPCS), reactor water cleanup (RWCU), and standby liquid control (SLC) piping system.

The licensee indicated that there are no changes in the reactor dome operating and design P-Ts, nor are there any changes in the MS operating and design P-Ts. There is a slight increase in the MS and FW flow rate and in the FW system operating P-T. The licensee reviewed the design basis parameters with regard to temperature, pressure, and flow rate for ASME Code stress calculation for the RCPB piping and its support components. The licensee found that the proposed power uprate conditions are bounded by the current licensing basis analysis of all piping systems. The licensee concluded that all safety aspects of the RCPB piping systems are adequate and acceptable for operation at the proposed 1.7 percent power uprate condition. Because there is no change in reactor dome operating and design P-Ts, the staff finds acceptable the licensee's conclusion that the design of piping, components, and their supports is adequate to maintain the structural and pressure boundary integrity of the reactor coolant loop for the proposed 1.7 percent power uprate condition.

3.2.4.2 Balance-of-Plant Piping and Safety-Related Valves

The licensee evaluated the balance-of-plant (BOP) piping systems by comparing the original design basis conditions with those for the proposed power uprate. In Reference 8.1d, the licensee indicated that the current licensing basis analyses for the BOP piping systems were performed at a reactor dome pressure of 1,074 psi absolute (psia) (versus the current licensed-pressure of 1,070 psia) and a core power level of 3,100 MWt (versus the proposed power level of 3,091 MWt). Therefore, the licensee indicated that the proposed power uprate conditions at 3,091 MWt or at 101.7 percent of the current rated power are enveloped by the current licensing basis.

The licensee also reviewed the piping stress analyses of record. The input parameters (temperature and pressure) used in the current BOP piping stress analyses remained bounding for the proposed power uprate. No new postulated pipe break locations were identified in any of the systems evaluated. The licensee concluded that the RBS BOP piping and related support systems remain within allowable stress limits in accordance with ASME Section III, 1974 Edition through the Summer 1976 addenda and American National Standards Institute (ANSI) B31.1 1973 Edition, as appropriate. Because temperature and pressure remain bounded by the analyses, the staff finds acceptable the licensee's conclusion that the BOP systems will operate at the proposed 1.7 percent power uprate conditions without adverse effects on the piping system and its supports.

As indicated by the licensee in Reference 8.1b, there is no change in the nominal vessel dome pressure. Also, the existing high energy line break (HELB) analyses were performed assuming 102 percent of the current power level, which bounds the proposed 1.7 percent power uprate condition. Therefore, the licensee concluded that the existing HELB analysis, break locations, pipe whip, and jet impingement analyses remain unchanged. The existing pipe whip restraints, jet impingement shields, and their supporting structures are also adequate for the proposed 1.7 percent power uprate condition. Because the current analyses remain bounding, the staff finds the licensee's conclusion acceptable.

The licensee reviewed its motor-operated-valve (MOV) program and indicated that the existing MOV evaluation at RBS was performed based on 102 percent of the current power level using maximum expected differential pressure, which bounds the proposed 1.7 percent power uprate condition. The licensee evaluated its commitments related to Generic Letter (GL) 95-07 (Reference 8.11), associated with the pressure locking and thermal binding of safety-related, power-operated gate valves that are required to operate to perform their intended safety functions. The licensee found that the existing analysis conditions remain bounding for the 1.7 percent power uprate. The licensee also evaluated its response related to GL 96-06 (Reference 8.12) regarding the over-pressurization of isolated piping segments. The licensee concluded that the existing evaluation for GL 96-06 was performed at 102 percent power and is, therefore, bounding for the proposed power uprate of 101.7 percent current rated power. Because current analyses remain bounding for the proposed uprated power conditions, the staff finds acceptable the licensee's conclusions that the power uprate will have no adverse effects on the safety-related valves, and that the licensee's conclusions from References 8.11 and 8.12, and GL 89-10 (Reference 8.13) regarding safety-related MOV testing and surveillance programs remain acceptable.

3.2.4.3 <u>Flow-Accelerated Corrosion in Piping</u>

The licensee stated that the carbon steel MS piping, carbon steel FW piping, and carbon steel high energy piping systems can be affected by flow accelerated corrosion (FAC). FAC is influenced by changes in fluid velocity, temperature, and moisture content which will result from the proposed power uprate condition. RBS has established a program for monitoring pipe wall thinning in single- and two-phase high energy flow carbon steel piping. The changes resulting from the power uprate condition have been evaluated and resulted in minor changes to the parameters affecting these systems. The licensee's continuing inspection program takes into consideration adjustments to predict material loss rates used to project the need for pipe maintenance/replacement prior to reaching minimum wall thickness requirements.

The staff requested the licensee to discuss the exclusion of the following systems from a table in Attachment 2 to Reference 8.1 which summarizes the evaluation of piping inside containment: recirculation system, RPV bottom head drain line, RHR, LPCS, HPCS and the RWCU. In Reference 8.1e, the licensee stated that the piping in the reactor recirculation, HPCS, and LPCS systems has been excluded from the FAC program based on industry guidance which takes into account steam quality, temperature, usage, fluid type, and pipe material content. In addition, the recirculation system is stainless steel and the core spray systems are both low temperature and low usage systems. The licensee also responded that the RHR and RWCU system piping are monitored by the FAC program. The licensee further clarified in its response that the most susceptible components to FAC are listed in the referenced table and are based on the CHECKWORKS predictive computer code. The staff finds this response acceptable since it is based on appropriate guidance, and accounts for specific design criteria which preclude FAC.

The staff requested the licensee to discuss the degree to which this program conforms with GL 89-08 (Reference 8.14). In its response, the licensee stated that the plant FAC program was described in its response to Reference 8.14, and was developed in accordance with Electric Power Research Institute and industry guidance. In addition, the program identifies the piping components and locations to be monitored, the acceptance criteria for these locations and components, and the corrective actions to be taken should these acceptance criteria not be met. This program will include appropriate changes to the piping inspection frequency to ensure adequate margin for those systems with changing process conditions. The staff finds this response acceptable since the FAC program conforms with the recommendations provided in Reference 8.14.

The staff requested the licensee to discuss the criteria considered in the selection of components inspected in the FAC program to demonstrate the program's effectiveness. The licensee responded that the following inputs are considered: components from the CHECKWORKS model, components from the "susceptible-not modeled" analysis, re-inspections based on the results of prior inspections, industry experience, and plant experience. The staff finds this response acceptable because these are appropriate criteria to consider in a robust FAC program.

On the basis of the information the licensee provided, the staff concludes that the proposed power uprate is acceptable with respect to FAC because the effects of this power uprate condition are adequately addressed in this program and will result in negligible effects on FAC.

3.2.5 Reactor Recirculation System

The power uprate will be accomplished by operating along extensions of the rod and core flow lines on the power/flow map. RBS is currently licensed to operate at up to a maximum core flow of 107 percent of the rated flow or 90.4 Mlb/hr. The power uprate does not require an increase in the maximum allowable core flow. Therefore, the reactor recirculation flow will be maintained within the flow limits of the existing power/flow map, with 100 percent power corresponding to the uprated power level. The cycle-specific reload analysis will consider the full range of the power and flow operating region.

The staff finds that the changes associated with the 1.7 percent power uprate will have an insignificant impact on the function of the recirculation system.

3.2.6 Main Steam Isolation Valves and Main Steamline Flow Restrictors

The MS isolation valves (MSIVs) are part of the RCPB and must be able to close within specific limits at all design and operating conditions upon receipt of a closure signal. The licensee states that the requirements for the MSIVs remain unchanged for the 1.7 percent power uprate, and that all safety and operational aspects of the MSIVs are within previous analyses performed at 102 percent of the CLTP. Regarding the MS line flow restrictors, the licensee states that the requirements for them remain unchanged for the power uprate because no change in steam break flow occurs (since the operating pressure is unchanged), and all safety and operational aspects of the flow restrictors are within previous analyses performed at 102 percent of the CLTP.

Based on the NRC staff review and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that predicted MSIV and MS line flow restrictor operation is bounded by current analyses, and plant operations at the 1.7 percent power uprate condition will have an insignificant impact on the ability of the MSIVs and MS line flow restrictors to meet their design objectives.

3.2.7 Reactor Core Isolation Cooling System

The generic discussion provided in Section 5.6.7 of Reference 8.4 is applicable to RBS.

The RCIC system provides core cooling when the RPV is isolated from the main condenser and the RPV pressure is greater than the maximum allowable for starting a low-pressure core cooling system. The RCIC system is designed to provide rated flow over a range of reactor pressures from 150 psi gauge (psig) to the maximum pressure corresponding to the lowest opening setpoint for the SRVs. In particular, the loss-of-FW (LOFW) flow transient assumes that the RCIC will maintain sufficient water level inside the core shroud high enough to ensure that the top of the active fuel will be covered throughout the event. The transient analysis also assumes that the low-setpoint SRVs will remove the stored and decay heat, since MSIV closure on low water level isolates the reactor from the main condenser. The transient is a power-dependent transient and is more severe at a higher initial power, since there is more stored energy and decay heat to be dissipated and the water level drops faster. The generic LOFW analysis described in NEDC-31984P (Reference 8.15) is applicable to RBS. The SRVs have capacity sufficient to perform their safety function at the proposed uprated power.

Since the proposed 1.7 percent power uprate does not increase the steady-state operating pressure or the SRV actuation setpoints, the staff finds that the RCIC performance would not be affected.

3.2.8 Residual Heat Removal System

The generic discussions provided in Section 5.6.4 of Reference 8.4 and Appendix J, Section 2.3.13 of the same report, are applicable to RBS.

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary-system decay heat removal after reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low-pressure coolant injection (LPCI) mode, the shutdown cooling mode, the suppression pool cooling mode, and the containment spray cooling mode.

The slightly higher decay heat has a negligible effect on the operation of the RHR system in the shutdown cooling mode. Because decay heat is only slightly increased under the proposed power uprate conditions and based on the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the RHR system is able to perform its required safety functions at the 1.7 percent power uprate condition.

3.2.9 Reactor Water Cleanup System

The primary parameters that affect the RWCU system are power transients, RWCU operating temperature and pressure, recirculation flow temperature, and system impurities such as fission and corrosion products. Power transients are the primary challenge to the RWCU system and are independent of the power uprate. The licensee stated that there is no significant effect on operating temperature and pressure conditions in the high-pressure portion of the system.

On the basis of the information the licensee provided, the staff concludes that the proposed power uprate is acceptable with respect to the RWCU system because the effects of this condition will not significantly affect the water chemistry performance requirements of the RWCU.

3.3 Engineered Safety Features

3.3.1 Containment System Performance

The containment system is designed to prevent the release of fission products to the environment in excess of that specified in 10 CFR Part 100, in the event of a design-basis accident (DBA). The license amendment request states that the previous containment evaluations are bounding for the 1.7 percent power uprate because they were performed at 102 percent of the CLTP. Although the nominal operating conditions increase slightly because of the power uprate, the required initial conditions for containment analysis inputs remain the same. The licensee's review included the short-term P-T response of the containment, the long-term temperature response of the suppression pool, the containment dynamic loads, the alternate shutdown cooling transient event, and containment isolation.

Because the initial conditions for containment analysis remain unchanged, and based on the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the containment system performance will not be affected by the 1.7 percent power uprate.

3.3.2 Emergency Core Cooling System

The ECCS is designed to provide protection in the event of a LOCA due to a rupture of the primary-system piping. Although DBAs are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on (1) the peak cladding temperature (PCT), (2) local cladding oxidation, (3) total hydrogen generation, (4) coolable core geometry, and (5) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential double-ended rupture of the largest recirculation system pipe. Assuming a single failure of the ECCS, the LOCA analyses identify the break sizes that most severely challenge the ECCS systems and the primary containment. The MAPLHGR operating limit is based on the most limiting LOCA analysis, and the licensees perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS for RBS includes the HPCS system, the LPCI mode of the RHR system, the LPCS system, and the automatic depressurization system (ADS).

3.3.2.1 High-Pressure Core Spray System

The HPCS system (with other ECCS systems as backups) is designed to maintain reactor water inventory during small-and intermediate-break LOCAs, isolation transients, and LOFW. The HPCS system is designed to pump water into the reactor vessel over a wide range of reactor operating pressures. The HPCS system also serves as a backup to the RCIC system. The system is designed to operate from normal offsite auxiliary power or from its dedicated emergency diesel generator (EDG).

The HPCS system is required to start and operate reliably over its design operating range. During the LOFW event and isolation transients, the RCIC maintains water level above the top of active fuel (TAF). For the MSIV closure events, the SRVs open and close as required to control pressure and HPCS eventually restores water level.

The licensee evaluated the capability of the HPCS system during operation at the TPO power level to provide core cooling to the reactor to prevent excessive fuel PCT following small- and intermediate-break LOCAs, and ensure core coverage up to the TAF in isolation transients and LOFW transients. The licensee stated that the HPCS evaluation is applicable to and is consistent with the evaluation in Section 5.6.7 of Reference 8.4. The maximum reactor pressure at which the HPCS system must be capable of injecting into the vessel for the RCIC backup function was selected based on the upper analytical values for the second lowest group of SRVs operating in the low set mode of operation. The TPO does not decrease the net positive suction head (NPSH) available for the HPCS pump or increase the required NPSH.

The licensee evaluated the capability of the HPCS system to perform as designed and analyzed its performance at the TPO conditions. The licensee determined that HPCS system can start and inject the required amount of coolant into the reactor for the range of reactor pressures associated with LOCAs and isolation transients. The TPO does not change the power required for the pump or the power required from the dedicated HPCS diesel generator.

Since the licensee's ECCS-LOCA analysis is based on the current HPCS capability (see Section 3.3.3 of this safety evaluation) and demonstrates that the system provides adequate core cooling, the staff finds the evaluation acceptable.

3.3.2.2 <u>Low-Pressure Core Spray System</u>

The LPCS system initiates automatically in the event of a LOCA. In conjunction with other ECCS systems, the core spray system provides adequate core cooling for all LOCA events. The system also provides spray cooling for long-term core cooling after a LOCA. The licensee explained that the existing LPCS system hardware has the capability to perform its design function at the TPO conditions and that the generic evaluation in Section 5.6.10 of Reference 8.4 is applicable to RBS.

Because the current ECCS-LOCA analysis demonstrates that the system provides adequate core cooling, the staff concludes that the LPCS is acceptable for TPO operation.

3.3.2.3 <u>Low-Pressure Coolant Injection Mode</u>

The LPCI system evaluation is applicable to and consistent with the evaluation in TLTR Section 5.6.10.

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. In conjunction with other ECCS systems, the LPCI mode is used to provide adequate core cooling for all LOCA events. The licensee stated that the existing system has the capability to perform the design injection function of the LPCI mode for operation at the TPO conditions.

Since the licensee's ECCS-LOCA analysis for TPO operation is based on the current LPCI capability (as discussed in Section 3.3.3 below) and demonstrates that the system provides adequate core cooling, the staff finds the evaluation acceptable.

3.3.2.4 Automatic Depressurization System

The ADS uses the SRVs to reduce reactor pressure after a small-break LOCA with HPCS failure, allowing LPCI and LPCS to provide cooling flow to the vessel. The plant design requires SRVs to have a minimum flow capacity. After a delay, the ADS actuates either on low water level plus high drywell pressure, or on low water level alone. The licensee stated that the ADS's ability to perform these functions is not affected by the power uprate. Since the small-break LOCA analyses assume that the ADS actuates at a bounding vessel pressure and power, the staff finds acceptable the licensee's assessment that the current power uprate does not affect the capability of the ADS to perform its function.

3.3.3 Emergency Core Cooling System Performance Evaluation

The ECCS is designed to provide protection against hypothetical LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The licensee stated that the ECCS performance under all LOCA conditions and the analytical models satisfy these requirements. The Framatome fuel was analyzed with Framatome's staff-approved RELAX and EXEM (Reference 8.6), and HUXY (Reference 8.16) models, while the GE fuel was analyzed with GE's NRC-approved SAFER/GESTAR model (Reference 8.5). These analyses were performed at 102 percent of current licensed power level for the power uprate. In both evaluations, the limiting case was the double-ended guillotine break of the recirculation line with failure of the HPCS system. Both of the analyses for each respective fuel type yielded PCTs less than 1,875 °F, peak metal water reactions less than 1 percent, and core-wide metal-water reactions less than 0.2 percent. These results comply with the 10 CFR 50.46 requirements of PCT of less than 2,200 °F, less than 17 percent cladding oxidation, and less than 1 percent core-wide metal-water reaction. The staff accepts Entergy's ECCS performance evaluation because the analytical models and codes are based on the NRC-approved methodology described in References 8.6 and 8.5, respectively, and the ECCS-LOCA analyses are based on bounding power and flow conditions.

The LOCA analyses of record demonstrate that the HPCS system, the LPCI mode of RHR, the LPCS system, and the ADS have the capabilities to provide core cooling during a LOCA. These capabilities do not change for operation at the uprated conditions. Therefore, the ECCS will continue to meet the ECCS-LOCA analysis assumptions and design criteria at the uprated conditions.

The staff finds the licensee's ECCS performance evaluation acceptable because the analytical models and codes are based on the NRC-approved methodology described in References 8.5 and 8.6, and because the ECCS-LOCA analyses are based on bounding power and flow conditions.

3.3.4 Main Control Room Atmospheric Control System

The main control room atmospheric control system minimizes unfiltered in-leakage following a DBA. Habitability (including control room operator doses) following a postulated accident from the 1.7 percent power uprate condition is unchanged because the main control room atmospheric control system had previously been evaluated for accident conditions at 102 percent of the CLTP. This evaluation is bounding for the proposed 1.7 percent power uprate.

Based on the foregoing and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the licensee's existing analysis for the main control room atmospheric control system is bounding for the 1.7 percent power uprate.

3.3.5 Standby Gas Treatment System and Main Steam Positive Leakage Control System

The Standby Gas Treatment System (SGTS) minimizes the offsite and control room doses during venting and purging of the containment atmosphere under abnormal conditions. The current capacity of this system was selected to maintain the secondary containment at a slightly negative pressure under such conditions. The charcoal beds in this system can accommodate DBA conditions at 102 percent of the CLTP.

The MS Positive Leakage Control System (MS-PLCS) prevents the release of fission products, in the event of leakage, through the closed MSIVs and MS drain lines after a design-basis LOCA. The MS-PLCS is adequate for the proposed power uprate since the current evaluations have been performed at 102 percent of the CLTP.

Based on the foregoing and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the licensee's existing analysis for the SGTS and the MS-PLCS remains valid for the 1.7 percent power uprate.

3.3.6 Post Loss-of-Coolant Accident Combustible Gas Control System

Hydrogen recombiners are used following a LOCA to maintain containment atmosphere hydrogen levels below combustible levels. The metal available for reaction is unchanged by the 1.7 percent power uprate and the hydrogen production, due to radiolytic decomposition, is unchanged because the system was previously evaluated for accident conditions at 102 percent of the CLTP.

Based on the foregoing and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the licensee's existing analysis bounds the 1.7 percent power uprate, and the impact on the hydrogen recombiners is negligible.

3.4 Instrumentation and Controls

Nuclear power plants are licensed to operate at a specified power, which, at operating power levels, is indicated in the control room by neutron flux instrumentation that has been calibrated to correspond to core thermal power. Core thermal power is determined by a calculation of the heat balance across the core. The accuracy of this calculation depends primarily upon the accuracy of FW flow, FW enthalpy, and MS enthalpy measurements. Thus, an accurate measurement of FW flow and temperature is necessary for calibrating nuclear instrumentation to represent core thermal power.

The instrumentation for measuring FW flow rate typically consists of a venturi, an orifice plate, or a flow nozzle to generate a differential pressure proportional to the FW velocity in the pipe. Of these three differential pressure devices, a venturi meter is most widely used for FW flow rate measurement in nuclear power plants. The FW temperature is typically measured using resistance temperature detectors mounted in the pipe.

The major advantage of using a venturi flow meter is the relatively low head loss created as the FW passes through the device. The major disadvantage of the venturi flow meter is the effect of venturi fouling upon flow meter instrument accuracy. Fouling causes a venturi flow meter to indicate higher differential pressures for equivalent flow velocities, which results in an output signal representing a higher than actual flow rate. Since FW flow rate is directly proportional to calorimetric power, this error in FW flow rate measurement leads the plant operator to calibrate the nuclear instrumentation at a higher than actual core power.

Calibrating the nuclear instrumentation to indicate higher than actual core power is conservative with respect to reactor safety, but causes the licensee to generate electrical power proportionately lower when the plant is operated at its indicated thermal power rating.

To eliminate this effect of venturi fouling on reactor power operating limits, the venturi flow meter device must be removed, cleaned, and calibrated. The high cost of venturi flow meter calibration and the need to improve flow instrumentation accuracy prompted the nuclear industry to assess other flow measurement techniques. The industry found the LEFM, which implements a transit time methodology, to be a viable alternative.

The transit time methodology to measure fluid velocity and temperature is based on the principle that ultrasonic pulses transmitted into a fluid stream travel faster in the direction of the fluid flow than opposite the flow. The difference in the upstream and downstream traversing times is proportional to the fluid velocity in the pipe. The average of the upstream and downstream transit times is proportional to the mean temperature of the fluid in the pipe. The fluid density in the pipe may be obtained using the measured pressure and the mean fluid temperature as inputs to a table of thermodynamic properties for water.

The LEFM uses multiple acoustic paths, instead of a single diagonal path, so that velocities measured along each path can be numerically integrated over the pipe cross section to determine the mean fluid velocity in the pipe. This fluid velocity is multiplied by a velocity profile correction factor, the pipe cross section area, and the fluid density to determine the FW mass flow rate in the piping. The velocity profile correction factor is derived from calibration testing of the LEFM in a plant-specific piping model at a calibration laboratory.

Caldon LEFM ✓+™ is a digital system controlled by software that consists of an electronic cabinet and a measurement section, or a spool piece, permanently mounted in each of the FW pipes. The licensee will install the Caldon LEFM ✓+™ system at RBS to measure FW flow and temperature. The licensee referenced proprietary Topical Reports ER-80P and ER-157P in its submittal for the proposed power uprate.

Caldon first developed the LEFM Check™ (✓™) system as described in Topical Report ER-80P, and subsequently issued its supplemental Topical Report ER-157P on an improved design, the LEFM ✓+™ system. Topical Report ER-80P describes the LEFM technology, includes calculations of power measurement uncertainty using an LEFM ✓™ system in a typical two-loop pressurized water reactor (PWR) or two-FW-line BWR, and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties. Supplemental Topical Report ER-157P describes the LEFM ✓+™ system and lists non-proprietary results of a typical PWR or BWR thermal power measurement uncertainty calculation using a single meter LEFM ✓™ or LEFM ✓+™ system (calculation with single meter measurement results in a bounding uncertainty value). These two reports, together, provide a generic basis and guidelines for power uprates using the Caldon LEFM ✓™ or LEFM ✓+™ system for FW flow and temperature measurements.

The LEFM ✓™ system uses eight transducers, two on each of the four acoustic paths in a single plane of the spool piece, where the velocity measured by any one of the four acoustic paths is the vector sum of the axial and the transverse components of fluid velocity as projected onto the path. The LEFM ✓+™ system uses sixteen transducers, eight each in two orthogonal planes of the spool piece. As such, the LEFM ✓+™ system is a combination of two LEFM ✓™ systems.

In the LEFM ✓+™ system, when the fluid velocity measured by an acoustic path in one plane is averaged with the fluid velocity measured by its companion path in the second plane, the transverse components of the two velocities are canceled and the average reflects only the axial velocity of the fluid. This makes the numerical integration of four pairs of averaged axial velocities and computation of volumetric flow inherently more accurate than can be obtained using four acoustic paths in a single plane. Also, since there are twice as many acoustic paths and there are two independent clocks to measure the transit time, errors due to uncertainties in path length and transit time measurements are reduced.

The licensee stated that the 1.7 percent power uprate is accomplished with no increase in the nominal vessel dome pressure. This minimizes changes to instrument setpoints related to system pressure. Satisfactory reactor pressure control capability is maintained by evaluating the steam flow margin available at the turbine inlet. The licensee stated that it will confirm this operational aspect of the power uprate by performing controller testing equivalent to the testing performed during the original startup of the plant which will provide additional assurance that reactor pressure will be appropriately controlled. The licensee included this commitment in Reference 8.1.

The licensee evaluated the instrumentation and control signal ranges and analytical limits for setpoints to determine the effects of the power uprate on process parameters. The licensee stated that setpoints were changed only to maintain adequate operating margins between plant operating parameters and trip values.

The staff safety evaluation on Caldon Topical Report ER-80P (Reference 8.17) included four additional items to be addressed by a licensee requesting a power uprate. The licensee submittal addressed each of the four items as follows:

(1) The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

The licensee stated that work will be controlled by procedures developed in accordance with Caldon recommendations. The incorporation of, and adherence to, these procedures will assure that the LEFM system is properly maintained and calibrated. The RBS instrumentation and control (I&C) personnel who perform initial maintenance on the LEFM system have been trained by Caldon, and the operation personnel will receive training on plant procedures affected by the power uprate.

The LEFM system features automatic self-checking and a continuously operating on-line test verifies that the digital circuits are operating correctly and within the specified accuracy envelope. If the LEFM system becomes inoperable, the control room operators are promptly alerted by control room indications.

In the event the LEFM system becomes inoperable, the RBS Technical Requirements Manual (TRM) will allow 72 hours operation at the uprate power level, provided steady state conditions persist during this period. If the plant experiences a reduction of power greater than 10 percent during the 72-hour Allowed Outage Time (AOT), the licensee will reduce operating power to the CLTP level of 3,039 MWt.

The FW venturis are continuously calibrated during normal operations using the flow measurement values provided by the LEFM. If the LEFM system becomes inoperable, the FW venturi measurements will be used in the plant heat balance calculations. The licensee's records indicate that a gradual degradation of the venturi accuracy, due to fouling and transmitter drift during a 72-hour AOT, translates into a maximum flow error of less than 0.009 percent of rated flow, and a resulting maximum error in core power measurement of less than 0.35 MW (0.01 percent of CLTP). These errors are within the margin of the licensee's core power measurement uncertainty calculations. The licensee concluded that there is no overall plant risk impact of continued operation at 3,091 MWt as a result of the change in FW venturi measurement uncertainty during the 72-hour AOT. The staff finds the licensee's justification for the 72-hour AOT consistent with Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," and NRC regulations and, therefore, is acceptable.

As stated above, in those instances in which (a) the LEFM system becomes inoperable or (b) there is a reduction of power greater than 10 percent while the LEFM is inoperable, the TRM imposes conditions on operation. Since the TRM is incorporated into the USAR, changes to the TRM must be controlled under the requirements of 10 CFR 50.59. Accordingly, these procedures will be subject to appropriate regulatory control.

(2) For plants that currently have LEFMs installed, the licensee should provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in topical report ER-80P.

The licensee stated that this criterion is not applicable to RBS because the LEFM has not been installed at RBS.

(3) The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current FW instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an

alternate methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installation for comparison.

The licensee provided the RBS power measurement uncertainty analysis supporting the RBS power uprate. The methodology is consistent with an NRC-approved GE method for calculating the MCPR safety limit uncertainties. The licensee used this process in its determination of the MCPR safety limit, as described in GE report NEDO-10958-A, “General Electric Thermal Analysis Basis” (Reference 8.19). NEDO-10958-A describes the GE Monte Carlo approach for generating the uncertainty in the core critical power ratio on the basis of uncertainties in BWR process variables.

The licensee used the RBS heat balance parameters and uncertainties to determine the core power uncertainty. With the exception of the recirculation pump motor efficiency, moisture carryover, CRD temperature, and thermal losses, each of the parameters is monitored by plant instrumentation, and was therefore modeled as an independent variable in the core power uncertainty calculation. Each parameter was varied in a normal distribution with the uncertainty reported in the GE NEDC-33051P (Attachment 1 to Reference 8.1e) to generate the core power uncertainty. The licensee used one million trials in the Monte Carlo approach, with each trial using randomly generated inputs to the heat balance equation to calculate the core power uncertainty.

For those variables that were not monitored with instrumentation (i.e., recirculation pump motor efficiency, moisture carryover, CRD temperature, and thermal losses), the licensee used conservative bounding values in the heat balance calculations. The staff reviewed the bounding values and found that the bounding values used by the licensee were conservative and, therefore, are acceptable.

The RBS heat balance calculation applies correlations to the steam tables to calculate the enthalpy at different pressures and temperatures for various heat balance inputs and outputs. As fits to the steam tables, these correlations may slightly deviate from the steam table values in some cases. Consequently, the licensee also applied an uncertainty to the enthalpy correlation applied in the core power uncertainty evaluation.

The licensee determined that the gross power uncertainty and the 2-sigma core power uncertainty is less than 11 MW to ensure that 102 percent CLTP would not be exceeded at the 95 percent probability and 95 percent confidence interval. To confirm this approach, the licensee applied the methodology reported in NUREG/CR-3659, “A Mathematical Model for Assessing the Uncertainties of Instrumentation Measurements for Power and Flow of PWR Reactors,” (Reference 8.20) using the same RBS input variable values as those used in the Monte Carlo analysis. (While Reference 8.20 was written for PWRs, the methodology and uncertainty values provided by the report are also applicable to BWR conditions.) The confirmatory calculation resulted in a 2-sigma power uncertainty of 9.77 MW. This value compares well with the Monte Carlo approach (~11 MW).

The licensee’s calculations confirm that the RBS core power measurement uncertainty for the proposed power uprate to 3,091 MWt will not exceed the Appendix K required core power limit. The staff reviewed the licensee’s analysis and finds the scope of the analysis is appropriate for the plant conditions at the uprated power level. The results of the uncertainty analysis show that the plant will continue to operate with an acceptable margin between the setpoints and the analytical limit.

(4) Licensees for plant installations where the ultrasonic meter (including the LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant- specific installation), should provide additional justification for use. This justification should show either that the meter installation is independent of the plant-specific flow profile for the stated accuracy or that the installation can be shown to be equivalent to known calibrations and the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

The licensee stated that this criterion is not applicable to RBS. The calibration factor for the RBS spool pieces will be established by test at Alden Research Laboratory, and then will be incorporated into the plant operating parameters. This action is acceptable.

The staff finds that the licensee’s responses to the four criteria in the staff safety evaluation sufficiently resolve the plant-specific concerns regarding maintenance and calibration of the LEFM system and other instrumentation affecting heat balance, hydraulic configuration of the installed LEFM, processes and contingencies for an inoperable LEFM, and methodology for calculating the LEFM and plant core power measurement uncertainties.

With regard to instrumentation and controls issues, on the basis of the above regulatory and technical evaluations of the licensee’s justifications for TS changes, the staff concludes that the licensee’s proposed power uprate is acceptable.

3.5 <u>Electrical Systems</u>

GDC 17, “Electric Power Systems,” of Appendix A to 10 CFR Part 50 requires that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure, among other things, that containment integrity and other vital functions are maintained in the event of postulated accidents.

As required by 10 CFR 50.63, “Loss of all alternating current power,” all light-water-cooled nuclear power plants must have the capability to withstand station blackout, as defined in 10 CFR 50.2, for an established period of time, and to recover therefrom.

Section 50.49 of 10 CFR Part 50, “Environmental qualification of electric equipment important to safety for nuclear power plants,” requires licensees to establish programs to qualify electric equipment important to safety. Under the rule, each licensee must prepare and maintain a record of qualification to document that each item of equipment subject to the rule (1) is qualified for its application, and (2) meets its specified performance requirements when subjected to the environmental conditions predicted to be present when it must perform its safety function up to the end of qualified life.

As described in Reference 8.1, the main` generator is rated at 1151megavolt-amps (MVA) at a 0.91 power factor. The station output generated at 22 kilo-Volts (kV), is fed through an isolated phase bus to the primary windings of the main power transformer connected to the 230 kV switchyard. The offsite power system consists of a 230 kV switchyard and an adjacent 500 kV switchyard interconnected by two banks of three single-phase transformers. Offsite power is supplied through four preferred station service transformers. Two of these transformers supply the non-safety-related 13.8 kV buses, and the remaining two transformers supply both the safety and non-safety 4.16 kV buses. The four preferred station service transformers are serviced by two 230 kV lines from the 230 kV switchyard. The Division I Class 1E bus is connected to a preferred station service transformer, and the Division II Class 1E bus is connected to the other preferred station service transformer. The offsite source to Division III (HPCS) Class 1E bus is taken from a non-Class 1E bus. The electrical distribution system has been previously evaluated to conform to GDC 17. Also, the plant has been previously evaluated for environmental qualification for electrical equipment per 10 CFR 50.49, and station blackout (SBO) per 10 CFR 50.63.

The staff's evaluation of the evaluation of grid stability, the main generator, the transformers, the EDGs, SBO, and environmental qualification is discussed in the following sections.

3.5.1 Grid Stability

The licensee performed a grid stability study in April 2000, using the 2002 summer peak as the maximum loading condition for load growth. The exciter and governor models were unaffected. The steady state analysis shows that the power uprate has little impact on grid stability. The existing grid stability analysis is applicable and bounding for the current 1.7 percent uprate request. There are no modifications associated with the power uprate, which would increase electrical loads significantly beyond those levels previously included in the distribution system.

The NRC staff reviewed the licensee’s submittal and concluded that the power uprate, because it is sufficiently small, will have no significant impact on grid stability. Therefore, the NRC staff concludes that the plant continues to have acceptable grid stability for this power uprate.

3.5.2 Main Generator

The main generator is rated at 1151 MVA at a 0.91 power factor. The power uprate of 1.7 percent does not affect the generator auxiliaries listed below since the generator will continue to operate below its design rating of 1151 MVA.

- Hydrogen gas system
- Primary water system
- Seal oil system
- Excitation system

The main generator performance is bounded by existing design and is not impacted by the power uprate. There is no impact of the power uprate on the protective relay settings of the main generator.

The staff reviewed the licensee’s submittal and concluded that the requested maximum power level is below the maximum main generator design rating of 1151 MVA and, therefore, operating the main generator at the uprated power condition is acceptable.

3.5.3 Main Transformer

The two main transformers are each rated for 518.6 MVA nameplate with a maximum rating of 788.5 MVA with additional cooling (forced oil and air (FOA)) installed. The total rating of the main transformers is 1577 MVA which is adequate for the power uprated condition.

The staff reviewed the licensee's submittal and concluded that the requested maximum power level is below the maximum main transformers design rating of 1577 MVA and, therefore, operating the main transformers at the uprated power condition is acceptable.

3.5.4 Isophase Bus

The isophase bus duct connects the main generator to the primary windings of the main transformer and the normal station service transformers. The isophase bus duct and the associated cooling equipment have adequate capacity for the proposed power uprate.

The staff reviewed the licensee's submittal and determined that operation at the 1.7 percent uprated power would be within the design rating of the isophase bus and, therefore, operation at the uprated power condition is acceptable.

3.5.5 Preferred Station Service Transformers

The preferred station service transformers 1E and 1F are rated at 230-13.8 kV, 51/68/85 MVA with OA (oil cooled by air)/FOA/FOA cooling. The preferred station service transformers 1C and 1D are rated at 230-4.6 kV, 10/12.5 MVA with OA/FA [forced air] cooling. Operation at the power uprated condition has no impact on the majority of electrical loads with the exception of condensate and FW pumps. The preferred station service transformers design ratings bound any expected bus loading increases.

The staff reviewed the licensee's submittal and determined that the increase in house loads resulting from the 1.7 percent power uprate, together with the existing house loads, remain within the maximum preferred station service transformers design rating and, therefore, operation at the uprated power condition is acceptable. Each of the service transformers is rated at 168 MVA at a 65°C FOA, which is well above the total station load of approximately 84 MVA. The associated cooling equipment will also support the power uprate for continuous operation with no modifications.

3.5.6 Normal Station Service Transformers

The normal station service transformers 1A and 1B are rated at 22-13.8 kV, 47.5 MVA with FOA cooling. The normal station service transformer 1C is rated at 22-4.16 kV, 16 MVA with FOA cooling. Operation at the power uprated condition has no impact on the majority of electrical loads with the exception of condensate and FW pumps. The normal station service transformers design ratings bound any expected bus loading increases.

The staff reviewed the licensee's submittal and concluded that the increase in house loads resulting from the 1.7 percent power uprate, together with existing house loads, remain within the maximum normal station service transformers design rating and, therefore, operation at the uprated power condition is acceptable.

3.5.7 Emergency Diesel Generators

The EDGs supply power following a loss of offsite power or degraded voltage conditions. The EDGs automatically supply alternating current power to the Class 1E buses in order to provide motive and control power to equipment required for a safe shutdown of the plant and mitigation and control of accidents. Station loads under emergency operation and distribution conditions are based on brake horse power (BHP). Emergency operation at the power uprated level is achieved by utilizing existing equipment operating at or below their nameplate ratings and within the calculated BHP. Therefore, under emergency conditions, the electrical supply and distribution components are adequately sized.

Accordingly, the staff's review determined that the power uprate does not affect the loading on the EDG and, therefore, the licensee will continue to meet GDC 17 requirements with the power uprate.

3.5.8 Direct Current Power

The direct current (dc) loading requirements were reviewed and no reactor power-dependent loads were identified. Operation at the uprated power level does not increase any loads or revise control logic. Therefore, there are no changes to the load, voltage drop, or short circuit current values.

The staff's review determined that the power uprate does not affect the loading on the dc power system and, therefore, the licensee will continue to meet GDC 17 requirements with the power uprate.

3.5.9 Environmental Qualification of Electrical Equipment

Conservatism in accordance with Institute of Electrical and Electronics Engineers (IEEE) 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Generating Stations" (Reference 8.18), was applied to the environmental parameters applicable to electrical equipment subject to the requirements of 10 CFR 50.49, and no change was needed for the power uprate. The licensee evaluated the equipment qualification for electrical equipment subject to 10 CFR 50.49 located inside the containment based on a DBA, MS line break (MSLB) and LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environments expected to exist during normal plant operation. Normal temperature increases remain less than conditions assumed in qualifying the equipment due to existing margins between such conditions and actual conditions and existing design margins in the ventilation systems. The current radiation levels under normal plant conditions increases slightly. The current environmental envelope for radiation is not exceeded by the changes resulting from the power uprate.

The licensee evaluated EQ (environmental qualification) for safety-related electrical equipment located outside the containment based on a MSLB in the pipe tunnel, or other HELBs, whichever is limiting for each plant area. The accident temperature, pressure, and humidity conditions resulting from a LOCA or HELB do not change with the uprated power level, but some HELB pressure profiles increase by a small amount. However, there is adequate margin in the qualification envelops to accommodate the small changes. Maximum accident radiation levels used for qualification of equipment outside containment are for existing analyses that bound the power uprated condition.

The NRC staff reviewed the licensee's submittal and determined that qualification data for the CLTP are bounding for the proposed uprate to 3,091 MWt; therefore, no changes to the EQ program are required for this power uprate, and the plant continues to meet the requirements of 10 CFR 50.49.

3.6 Auxiliary Systems

3.6.1 Fuel Pool - Cooling and Design

The fuel pool cooling and cleanup system (FPCCS) removes heat from the spent fuel assemblies stored in the spent fuel pool in order to maintain the pool temperature at or below its design temperature during normal plant operations. In addition, the FPCCS reduces activity, maintains water clarity, and maintains the cooling function during and after a seismic event.

The fuel pool heat load increases slightly as a result of the power uprate. However, the new heat load is within the design-basis heat load for the FPCCS, and will not result in a delay in removing the RHR system from service when fuel is removed from the reactor and placed in the pool (i.e., the duration of supplemental cooling for the upper containment fuel pool will not be increased). The licensee determined that the spent fuel pool cooling is adequate by calculating the heat load generated by a full core discharge plus remaining spaces filled with used fuel discharged at regular intervals.

Regarding other fuel pool design considerations, the crud activity and corrosion products in the spent fuel pool can increase slightly; however, this increase is insignificant and the water quality will be maintained by the FPCCS. In addition, the normal radiation levels around the spent fuel pool may increase slightly; however, the operational doses to personnel or equipment will not significantly increase. Also, there is no effect on the design of the spent fuel racks because the original spent fuel pool design temperature is not exceeded.

Based on the above considerations and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the FPCCS, in combination with the RHR system, can maintain the spent fuel pool temperature at or below design limits for all core offload conditions at the proposed 1.7 percent uprated power level.

3.6.2 Water Systems

The safety-related standby service water (SSW) system provides cooling water during and following a DBA. The safety-related performance of the SSW system during and following the most demanding design-basis event (LOCA) does not change because the original LOCA analysis was based on 102 percent of the CLTP. Similarly, the containment response analysis is also based on 102 percent of the CLTP. In addition, there is no change in the safety-related heat loads and the existing capacity of the RHR and SSW systems are adequate.

Regarding non-safety-related heat loads, the major service water heat load increases from the 1.7 percent power uprate are due to increases in main generator losses rejected to the stator water coolers and hydrogen coolers and the turbine plant component cooling water (TPCCW) system. The increase in service water heat load from these sources is proportional to the 1.7 percent power uprate; however, the design is adequate to handle the power uprate.

The main condenser, circulating water, and normal heat sink systems are designed to remove the heat rejected to the condenser and maintain a low condenser pressure. The 1.7 percent power uprate increases the heat rejected to the condenser and may reduce the difference between the operating pressure and the required minimum condenser vacuum; however, the designs of the condenser, circulating water system, and heat sink are adequate for the power uprate.

The heat loads on the reactor plant component cooling water (RPCCW) system do not increase significantly, due to the 1.7 percent power uprate, because they depend on either reactor vessel water temperature or flow rates in the systems cooled by the RPCCW. The change in reactor vessel water temperature is minimal and there is no change in nominal reactor operating pressure. The RPCCW system will experience a slight heat load increase, primarily in the fuel pool cooler heat exchangers; however, the RPCCW system has adequate design margin to remove the additional heat load.

The power-dependent heat loads on the TPCCW system, which increase due to the 1.7 percent power uprate, are those related to the operation of the bus duct cooler and exciter coolers. The remaining heat loads are not strongly dependent on reactor power and do not increase significantly. The TPCCW has sufficient capacity to remove the additional heat load.

The ultimate heat sink (UHS) is provided by the SSW cooling tower. The post-LOCA UHS water temperature will increase slightly due to the 1.7 percent power uprate, primarily due to higher reactor decay heat. However, the ability of the UHS to perform required safety functions was demonstrated by the licensee with previous analyses based on 102 percent of the CLTP.

Based on the foregoing and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that plant operations at the proposed 1.7 percent uprated power level do not change the design aspects and operations of the water systems. Therefore, the staff finds that the impact of plant operations at the proposed uprated power level on these systems is acceptable.

3.6.3 Standby Liquid Control System

The SLC system provides an alternate means of attaining and maintaining cold shutdown conditions, assuming no control rod movement, as required by GDC 26, "Reactivity Control System Redundancy and Capability."

The shutdown capability of the SLC system and the boron solution necessary are evaluated each reload cycle. Since the SRV setpoints are not changed for the proposed power uprate, the uprate will have no effect on the rated injection flow. The licensee determined that the capability of the SLC system to provide its backup shutdown function is unchanged and it will continue to meet the requirements of 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." Because the uprate will not change the operating parameters of the SLC system, the staff agrees that the SLC will perform acceptably during TPO operation.

3.6.4 Heating, Ventilation, and Air Conditioning Systems

The function of the Heating, Ventilation, and Air Conditioning (HVAC) systems is to prevent extreme thermal environmental conditions from impacting personnel and equipment by ensuring that design temperatures are not exceeded. HVAC systems that could potentially be affected by the requested power uprate include heating, cooling, exhaust, and recirculation units in the turbine building, containment building, and the drywell, auxiliary building, fuel handling building, control building, and the radwaste building.

The 1.7 percent power uprate results in a minor increase in heat load caused by the slightly higher FW process temperature. The increased heat load is within the margin of the steam tunnel area coolers. In the drywell, the increased heat load due to the FW process temperature is within the cooling system capacity. In the turbine building, the maximum temperature increases in the FW heater bay, and condenser areas are less than 2 °F due to the increase in the FW process temperatures. In the fuel building, the increase in heat load due to a slight spent fuel pool cooling process temperature increase is within the margin of the area coolers. Other areas are unaffected by the power uprate because the process temperatures and electrical heat loads remain constant.

Based on the foregoing and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that plant operation at the proposed uprated power level will have an insignificant or no impact on the HVAC systems for the above-cited areas.

3.6.5 Fire Protection and 10 CFR Part 50, Appendix R

Fire suppression or detection is not expected to be impacted by plant operations at the proposed 1.7 percent uprated power level since there are no physical plant configuration changes or combustible load changes resulting from the uprated power operations. In addition, the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions are adequate and do not change, and the operator actions required to mitigate the consequences of a fire are not affected by the uprated conditions.

The RBS Appendix R fire event analysis assumes an operating power level of 102 percent of the CLTP at the start of the fire event, which bounds the 1.7 percent power uprate conditions. The power uprate does not cause an increase in peak vessel bottom pressure, maximum containment pressure, or maximum containment temperature. In addition, the PCT remains well below 1,500 °F.

Based on the foregoing and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the safe shutdown systems and procedures used to mitigate the consequences of a fire will continue to meet 10 CFR 50.48, "Fire protection," and 10 CFR Part 50, Appendix R, and will not be affected by plant operations at the proposed 1.7 percent power uprate.

3.7 Power Conversion Systems

The turbine generator is designed with a maximum flow-passing and generator capability in excess of rated conditions to ensure that the design rated output is achieved. Steam specific calculations were performed to determine the turbine steam path conditions for the 1.7 percent power uprate. These operating conditions are bounded by previous analysis of the turbine and generator stationary and rotating components. In addition, valves, control systems, and other support systems are bounded by the existing analyses. The existing rotor missile analysis was performed at conditions that bound the power uprate.

The condenser capability was evaluated for performance at the 1.7 percent power uprated conditions based on current circulating water system flow. The design margin in the condenser heat removal capability can accommodate the additional heat rejected for operation at the uprated power condition. Regarding the steam jet air ejectors, air leakage into the condenser does not increase as a result of the 1.7 percent power uprate, and the small increase in hydrogen and oxygen flows from the reactor does not affect the steam jet air ejector performance because the design was based on operation at significantly greater than required flows.

The steam bypass pressure control system has a steam bypass capacity which is greater than or equal to 9.5 percent of the uprated rated-thermal-power steam flow rate. The transient analysis that credits the turbine bypass system uses a bypass capacity that is less than the actual capacity. In addition, this system is not safety-related.

The FW and condensate systems are not safety-related; however, their performance may have an affect on plant operation at the 1.7 percent uprated condition. The FW components are capable of providing the slightly higher uprated FW flow rate at the desired temperature and pressure, and the FW control valves are capable of maintaining water level control at the uprated conditions. The condensate demineralizers will experience slightly higher loadings at the uprated condition, which will result in slightly reduced run times. However, the reduced run times are acceptable because the licensee will monitor demineralizer performance to maintain condensate chemistry within the values set forth in Technical Requirements Manual section 3.4.13, Chemistry, and will replace demineralizer resin more frequently, if necessary.

Based on the foregoing and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the power conversion systems can accommodate plant operations at the proposed 1.7 percent uprated power level. Therefore, the staff finds that the impact of plant operations at the proposed uprated power level on these systems is acceptable.

3.8 Radwaste and Radiation Sources

Regarding liquid and solid waste management, the licensee states that the activated corrosion products in liquid wastes are expected to increase proportionally to the power uprate, and the total volume of processed waste is not expected to increase appreciably because the only significant increase in processed waste is due to the more frequent backwashes of the condensate demineralizers and the RWCU filter demineralizers. The licensee has reviewed the plant operating effluent reports and the slight increases expected from the 1.7 percent power uprate, and concluded that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, will continue to be met.

Regarding gaseous waste management, the licensee states that the activity of airborne effluents released through building vents is not expected to increase significantly with the 1.7 percent power uprate. The release limit is an administratively controlled variable, and is not a function of core power. In addition, gaseous effluents are expected to remain well within the release limits following the power uprate. Regarding core radiolysis (the formation of hydrogen and oxygen), which increases linearly with core power, the licensee states that the radiolytic hydrogen flow rate increases, but remains well within the design capacity of the offgas recombiner system. The licensee also states that the gaseous waste management systems, which include the offgas system and the various building ventilation systems, are designed to meet the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I.

Based on the foregoing and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, will continue to be satisfied by the solid, liquid, and gaseous radwaste management systems at the proposed 1.7 percent increase in power level.

3.9 Reactor Safety Performance Evaluation

3.9.1 Anticipated Operational Occurrences - Reactor Transients

AOOs are abnormal transients which are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, GDC 10, GDC 15, "Reactor Coolant System Design," and GDC 20, "Protection System Functions." GDC 10 requires that the reactor core and associated control and instrumentation systems be designed with sufficient margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation and during AOOs. GDC 15 requires that sufficient margin be included to ensure that the design conditions of the RCPB are not exceeded during normal operating conditions and AOOs. GDC 20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure that the specified fuel design limits are not exceeded during any normal operating condition and AOOs.

Reference 8.7 provides further guidelines: (1) pressure in the reactor coolant and MS system should be maintained below 110 percent of the design values according to the ASME Code, Section III, Article NB-7000, “Overpressure Protection;” (2) fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during normal operating conditions and AOOs; (3) an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and (4) an incident of moderate frequency, in combination with any single active-component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel cladding perforations are acceptable.

Chapter 15 of the RBS USAR contains analyses of the effects of an AOO resulting from changes in the system parameters such as (1) a decrease in core coolant temperature, (2) an increase in reactor pressure, (3) a decrease in reactor coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, and (6) a decrease in reactor coolant inventory. The facility’s responses to the most limiting transients are analyzed each reload cycle and corresponding changes in the MCPR are added to the SLMCPR to establish the operating limit MCPR. A potentially limiting event is an event or accident that has the potential to affect the core operating and safety limits.

Since the licensee will perform the reload analysis at the uprated conditions using an NRC-approved methodology and has determined that the thermal limits to ensure the fuel cladding integrity will be maintained for operation at the uprated conditions during AOOs and accidents, acceptable acceptance criteria are met.

3.9.2 Radiological Analysis of Design Basis Accidents

In addition to the material included in Reference 8.1, the staff reviewed relevant information in Chapter 15 of the RBS USAR.

The potential radiological consequences of DBAs are proportional to the quantity of radioactive material released to the environment. This release is a product of the radioactive material released from the core or from the reactor coolant system (RCS), the transport of the released material to the effluent release point, and the transport (e.g., atmospheric dispersion) in the environment. The transport in the environment is not considered further as it is not affected by the power uprate. In general, the inventory of fission products in the reactor core and the quantity of radioactive material in the RCS is directly proportional to the operating power level. An increase in RTP, as proposed by the licensee, can be expected to increase the inventory of radioactive material available for release. The transport of the released material is dependent on plant process parameters, such as process steam flows, temperatures, and pressures. An increase in the RTP and any associated plant modifications could affect the assumptions made in previous consequence analyses.

The staff reviewed EOI's submittal and Chapter 15 of the RBS USAR. The decrease in reactor power measurement uncertainty due to the installation of the LEFM effectively offsets the proposed increase of 1.7 percent in RTP. Staff review of the dose analyses in Chapter 15 of the RBS USAR indicated that the current analyzed power level bounds the requested uprate power level when the reduced power uncertainty associated with the LEFM is considered.

RBS USAR Chapter 15 addresses the consequences of postulated DBAs at RBS. For each analyzed accident in which a release of radionuclides is postulated, there is a discussion providing the methodology, assumptions, and inputs used in performing the current radiological consequence assessments. Those discussions and the associated tables indicate that the analyses were performed using a lower level equivalent to 102 percent of RTP (3100 MWt). Since the current mass and energy releases were determined using a power level equivalent to 102 percent of RTP or greater, the results of those analyses continue to bound the proposed 1.7 percent uprate plus 0.3 percent measurement uncertainty.

The staff concludes that there is reasonable assurance that operation at the increased RTP of 3,091 MWt will not result in postulated DBA radiological consequences that exceed the analysis results currently documented in the RBS USAR. Since these DBA analysis results were previously found to meet the acceptance criteria of 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC 19, "Control Room," the 1.7 percent power uprate coincident with a 1.7 percent decrease in flow measurement uncertainty and the proposed conforming TS changes are acceptable with regard to the DBA radiological consequences.

3.9.3 Special Events

3.9.3.1 Anticipated Transient Without Scram

ATWS is defined as an AOO with failure of the reactor protection system to initiate a reactor scram to terminate the event. The requirements for ATWS are specified in 10 CFR 50.62. The regulation requires BWR facilities to have the following mitigating features for an ATWS event:

(1) a SLC system with the capability of injecting a borated water solution with reactivity control equivalent to the control obtained by injecting 86 gpm of a 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside-diameter reactor vessel

(2) an Alternate Rod Injection (ARI) system that is designed to perform its function in a reliable manner and that is independent from sensor output to the final actuation device

(3) equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS

RBS meets the ATWS mitigation requirements defined in 10 CFR 50.62. RBS has an SLC system capable of boron injection equivalent to 86 gpm, and has installed an ARI system and an automatic Recirculating Pump Trip (RPT) logic.

BWR facilities are also analyzed against certain ATWS acceptance criteria to demonstrate the ability to withstand an ATWS event. These criteria include maintaining fuel integrity (the core and fuel must maintain a coolable geometry), primary system integrity (the peak reactor vessel pressure must remain below 1,500 psig), and containment integrity (the containment temperature and pressure must not exceed the design limit).

Section 5.3.5 and Appendix L of Reference 8.4 present a generic evaluation of BWRs to an ATWS event after a TPO uprate. Reference 8.4 provides an ATWS acceptance criterion; if the criterion is not met, then a plant-specific ATWS containment analysis is required. RBS has a 4.7 °F suppression pool temperature margin available, based on a GE core containment analysis performed at the current licensed RTP. Based on the generic evaluation and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the impact of the RBS mixed core on the peak suppression pool temperature for a 1.7 percent uprate is less than the current suppression pool temperature margin of 4.7°F.

Based on the criteria and justification provided in Reference 8.4 and the analyses performed by GE, as well as the available margin for peak ATWS parameters, the staff finds the licensee's evaluation acceptable. Accordingly, the staff concludes that RBS meets the ATWS rule requirements specified in 10 CFR 50.62.

3.9.3.2 Station Blackout

The licensee performed the SBO coping capability analysis prior to the recent power uprate submittal. The containment conditions evaluated for an SBO event remain bounding for the power uprated conditions. The auxiliary building conditions during an SBO were evaluated.

The input that was dependent upon reactor conditions was the heat load to the auxiliary building. The impact of the power uprate on the heat load was evaluated and it was concluded there was no significant impact of the heat release to the auxiliary building due to the following reasons:

(1) The piping heat release was calculated using the most conservative suppression pool temperature response, which bounds the temperatures calculated for the power uprate.

(2) RCIC steam piping heat loads were originally based upon a steam temperature of 575 °F, which bounded the power uprate steam temperature of 553 °F. Since the steam temperature is not changing for the power uprate, the existing heat loads are bounding.

The staff reviewed the licensee's submittal and determined that the postulated SBO scenarios for the power uprate are bounded by the current evaluations, therefore, the plant continues to meet the requirements of 10 CFR 50.63 and the design is acceptable.

3.10 Other Evaluations

3.10.1 High-Energy Line Break Analyses

The licensee stated that since the 1.7 percent power uprate system operating temperatures and pressures change only slightly, there is no significant change in HELB mass and energy release. Therefore, the consequences of any postulated HELB would not significantly change.

The licensee's HELB analysis addressed all systems evaluated in the USAR. At the uprated power level, HELBs outside the drywell would result in an insignificant change in the subcompartment pressure and temperature profiles. Its evaluation shows that the affected building and cubicles that support safety-related functions are designed to withstand the resulting pressure and thermal loading following an HELB at the uprated power level.

The postulated break locations remain the same because the piping configuration does not change due to the 1.7 percent power uprate. In addition, the existing calculations for the development of pipe whip and jet impingement loads from the postulated HELBs have been determined to be bounding for the safe shutdown of the plant for the 1.7 percent power uprate. Therefore, the existing pipe whip restraints and jet impingement shields are adequate.

Based on the foregoing and the experience gained from the review of power uprate applications for similar BWR plants, the staff finds that the consequences of any postulated HELB would not change significantly and will be acceptable for plant operations at the uprated power level.

3.11 Human Factors

3.11.1 Emergency and Abnormal Operating Procedures

The licensee stated that their design control program requires that the implementing procedures be identified during the modification development. The licensee has reviewed the list of procedures affected by the recent, previous 5 percent power uprate and considered the impact of this 1.7 percent power uprate on these procedures. The licensee indicated that operating procedures changes necessary to reflect operation at the 1.7 percent uprate power conditions are identified as part of the associated modification process, and that these procedures will be revised prior to operation at the increased power level.

The staff finds the licensee's response acceptable because the licensee has a process to identify the plant procedures that will be affected by the 1.7 percent power uprate and indicated that the procedures will be appropriately revised.

3.11.2 Risk-Important Operator Actions Sensitive to Power Uprate

The licensee stated that for the 1.7 percent power uprate conditions, operator response to transient, accident, and special events are not affected. Operator actions for maintaining safe shutdown, core cooling, containment cooling, etc., do not change for the power uprate.

The staff finds the licensee's response acceptable because the licensee has adequately addressed the question of operator actions sensitive to the power uprate by describing the lack of effect on operator performance and operator response.

3.11.3 Control Room Controls, Displays, and Alarms

The licensee stated that operator interface changes to support the 1.7 percent power uprate include the addition of computer points to provide indication of FW flow, FW temperature and FW pressure signals from the new, more accurate flow meter. (During a follow-up discussion with the licensee, the licensee indicated that these new computer points will be available for display on an existing computer screen in the control room, and if there is a failure of the new flow meter, this would be displayed by a scrolling message across the bottom of the screen.) The only setpoint changes will be to variables such as the turbine stop valve closure and the TCV fast closure scram bypass setpoints, and the rod pattern controller low and high power setpoints, which will remain the same in terms of absolute power level, but will be slightly lower in terms of percent of RTP. Minor changes to the TSs, power/flow map, flow-referenced setpoint, and the like, will be communicated through routine operator training prior to operation at the uprated power level.

The licensee stated that no changes to zone markings on control room meters are anticipated. In addition, no control room displays or controls will be upgraded from analog to digital.

The staff finds the licensee's response acceptable because the licensee has adequately identified the changes that will occur to controls, displays, and alarms as a result of the power uprate, and described how these changes will be accommodated.

3.11.4 Safety Parameter Display System

The licensee stated that the 1.7 percent power uprate will have negligible impact on the safety parameter display system (SPDS). The SPDS monitors and provides a status-board display of key parameters that are entry points into the emergency procedures. None of the entry conditions are affected by the 1.7 percent power uprate. All points remain within their existing ranges. Affected operating values, such as reactor coolant temperature and pressure, are addressed in the applicable operating procedures.

The staff finds the licensee's response acceptable because the licensee has adequately identified the changes that will occur to the SPDS as a result of the power uprate and described how the changes will be addressed.

3.11.5 Operator Training Program and the Control Room Simulator

Regarding the operator training program, the licensee stated that no additional training (apart from normal training) is required to operate the plant at the uprated condition. Minor changes to the TSs, power/flow map, flow-referenced setpoint, and the like, will be communicated through routine operator training prior to operation at the uprated power level.

Regarding the control room simulator, the licensee stated that no hardware changes have been identified to reflect the power uprate conditions. Simulator software changes and validation are controlled in accordance with ANSI/American Nuclear Society (ANS) 3.5-1998, "Nuclear Power Plant Simulators for Use in Operator Training and Examination."

Changes made to plant systems for the 1.7 percent power uprate will also be made in the simulator for those systems currently modeled. Simulator changes identified at this time include the turbine stop valve closure and the TCV fast closure scram bypass setpoints, the rod pattern controller low and high power setpoints, the 100 percent thermal power reference value, and the addition of computer points to reflect the new flow input. The licensee indicated that changes made to support this power uprate will be completed prior to operation above the CLTP level.

The staff finds the licensee's response acceptable because the licensee has adequately addressed the changes to the operator training program and how the simulator will accommodate the changes.

3.11.6 Summary - Human Performance

Based on the evaluation in Sections 3.11.1 through 3.11.5 of this safety evaluation, the staff concludes that the previously discussed review topics associated with the proposed power uprate have been satisfactorily addressed. The staff further concludes that the power uprate should not adversely affect simulation facility fidelity or operator performance.

3.12 Facility Operating License and Technical Specification Changes

The licensee proposed to revise the FOL and TSs as follows to reflect the increase in licensed power level from 3,039 MWt to 3,091 MWt:

(1) Paragraph 2.C.(1) of FOL NPF-47, “Maximum Power Level,” would be revised to authorize operation of the facility at reactor core power levels not in excess of 3,091 MWt (100 percent rated power).

(2) The definition of RTP in TS Section 1.1, “Definitions,” would be revised to state that the RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3,091 MWt.

(3) The statement of the Limiting Condition for Operation 3.4.1, Condition B.1 and Required Actions B and B.1 of TS 3.4.1 are revised to adjust the maximum thermal power for single loop operation from 79 percent to 77.6 percent RTP.

The FOL and TS changes reflect the proposed increase in licensed power level based on installation of the Caldon LEFM✓+™ System for FW flow and temperature measurements.

Based on the evaluations discussed in Sections 3.1 through 3.11 of this safety evaluation, the staff concludes that the above-described changes to the FOL and TSs are acceptable.

4.0 REGULATORY COMMITMENTS

The licensee included regulatory commitments in its application and its responses to the NRC staff RAIs. The commitments relevant to the NRC staff evaluations are listed in the following table.

Regulatory Commitments

	TYPE (Check One)		SCHEDULED COMPLETION DATE (If required)
COMMITMENT	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The following commitments are from Reference 8.1 :			
This operational aspect of the TPO uprate (steam flow margin available at the turbine inlet) will be demonstrated by performing controller testing equivalent to the testing performed during the original startup of the plant. (TSAR 1.2.1)	X		upon implementation
The values used in the measurement uncertainty calculation will be confirmed by the initial calibration test results of the LEFM CheckPlus system. (TSAR 1.4)	X		upon implementation
PCS [pressure control system] tests will be performed during the power ascension phase. (TSAR 5.2.1, 10.4)	X		upon implementation

COMMITMENT	TYPE (Check One)		SCHEDULED COMPLETION DATE (If required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The performance of the FW/level control systems will be recorded at 95% and 100% of CLTP and confirmed at the TPO RTP during power ascension. These checks will demonstrate acceptable operational capability and will utilize the methods and criteria described in the original startup testing of these systems. (TSAR 5.2.2)	X		upon implementation
The reload analyses performed prior to TPO implementation will be based on the reactor power bypass AL [analytical limit] for the TSV [turbine stop valve] closure scram. TCV [turbine control valve] fast closure scram, and RPT remaining constant in percent of RTP. (TSAR 5.3.2)	X		RF11
The measurements [of reactor and system pressure and flows] will be taken along the same rod pattern line used for the increase to TPO RTP. Core power from the APRMs [average power range monitors] is re-scaled to the TPO RTP before exceeding the CLTP and any necessary adjustments will be made to APRM alarm and trip settings. (TSAR 10.4)	X		upon implementation

COMMITMENT	TYPE (Check One) ONE-TIME ACTION	CONTINUING COMPLIANCE	SCHEDULED COMPLETION DATE (If required)
Demonstration of acceptable fuel thermal margin will be performed prior to and during power ascension to the TPO RTP at each steady-state heat balance point defined above. Fuel thermal margin will be projected to the TPO RTP point after the measurements taken at 100% of CLTP to show the estimated margin. The thermal margin will be confirmed by the measurements taken at full TPO RTP conditions. The demonstration of core and fuel conditions will be performed with the methods currently used at the plant. (TSAR 10.4)	X		upon implementation
The Emergency Operating Procedures (EOP) action thresholds are plant unique and will be addressed using standard procedure updating processes. (TSAR 10.8)	X		upon implementation
A non-proprietary version [of NEDC-33051] is planned and will be submitted by early July. (letter)	X		7/15/02
The installation of each flow element will conform to the requirements in Caldon Topical Reports ER-80P and ER-157P. (att 1, 4.2.1)	X		4/15/02
The administrative controls [for LEFM OOS [out of service]] will be added to the RBS Technical Requirements Manual. (att 1, 4.2.2)	X		upon implementation
Calibration and maintenance of the LEFM system will be performed using site procedures developed from the Caldon LEFM ✓+™ System technical manuals. (att 1, 4.2.3)	X		upon implementation
RBS I&C personnel performing initial maintenance on the LEFM will be trained by Caldon. (att. 1, 4.2.3)	X		upon implementation

COMMITMENT	TYPE (Check One)		SCHEDULED COMPLETION DATE (If required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Procedures governing normal operation, emergency operation, and off-normal operation that may be affected by the power uprate will be identified in the design change process and revised prior to implementation of the uprated power. Appropriate personnel will receive training on the LEFM ✓+™ System as well as on the affected procedures. This training is to consist of briefings, required reading, classroom sessions, and a simulator demonstration. (att 1, 4.2.4)	X		upon implementation
LEFM System operating procedures will ensure that the assumptions and requirements of the uncertainty calculation remain valid. (att 1, 4.2.5)	X		upon implementation
The LEFM spool pieces will be calibrated at Alden Research Laboratory (ARL). Profiles comparisons will be made between ARL and plant commissioning. Differences will be considered in the final overall plant calorimetric uncertainty analysis to ensure consistency with ER-80P and ER-157P. (att 1, 4.2.7)	X		3/14/03
This information [description of RBS uncertainty analysis and AOT basis] will be provided once the RBS uncertainty analysis has been verified. (att 1, 4.2.8, cr 3, & it I.E & I.G)	X		6/30/02
The contingent actions when an LEFM is out of service will be provided once the RBS uncertainty analysis is verified. (att 1, 4.2.8, it I.H; also 4.2.2)	X		6/30/02

COMMITMENT	TYPE (Check One)		SCHEDULED COMPLETION DATE (If required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The following commitments are from Reference 8.1d:			
Minor changes to the Technical Specifications, power/flow map, flow-referenced setpoint and the like will be communicated through routine operator training prior to operation at the uprated power level.	X		upon implementation
The procedures that impact plant operation have been identified and will be implemented prior to operation above CLTP.	X		upon implementation
Changes made to the simulator to support this power uprate project will be completed prior to operation above the current licensed thermal power level.	X		upon implementation

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Louisiana was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (67 FR 40022, published June 11, 2002), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 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.

8.0 REFERENCES

8.1 RBG-45951, Paul D. Hinnenkamp (Entergy) letter to NRC, "Appendix K Measurement Uncertainty Recovery - Power Uprate Request," dated May 14, 2002. (Attachment 2 was withdrawn and superceded by Reference 8.1e)

8.1a RBG-45974, Rick J. King (Entergy) letter to NRC, "Non-proprietary TSAR [Technical Specification Amendment Request] in Support of Appendix K Measurement Uncertainty Recovery - Power Uprate Request," dated July 9, 2002. (Attachment 1 superceded by Reference 8.1e)

8.1b RBG-45984, Rick J. King (Entergy) letter to NRC, "Response to Requests for Additional Information, Appendix K Measurement Uncertainty Recovery - Power Uprate Request," dated August 2, 2002.

8.1c RBG-46011, Rick J. King (Entergy) letter to NRC, "Response to Requests for Additional Information, Appendix K Measurement Uncertainty Recovery - Power Uprate Request," dated September 16, 2002.

8.1d RBG-46012, Rick J. King (Entergy) letter to NRC, "Response to Requests for Additional Information, Appendix K Measurement Uncertainty Recovery - Power Uprate Request," dated November 7, 2002.

8.1e RBG-46036, Michael A. Krupa (Entergy) letter to NRC, "Revision to TSAR in Support of Appendix K Measurement Uncertainty Recovery Power Uprate Request," dated November 22, 2002.

8.2 John N. Hannon (NRC) letter to C. L. Terry (TU Electric), "Staff Acceptance of Caldon Topical Report ER-80P: Improving Thermal Power Accuracy While Increasing Power Level Using The LEFM System," dated March 8, 1999.

8.3 Stuart A. Richards (NRC) letter to Michael A. Krupa (Entergy), "Review of Caldon, Inc. Engineering Report ER-157P," dated December 20, 2001.

8.4 General Electric, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization (TLTR)," Licensing Topical Report NEDC-32938P, July 2000.

8.5 General Electric, "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, July 2000.

8.6 Siemens Power Corporation, "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," ANF-91-048(P)(A), January 1993, Supplements 1 and 2, "BWR Jet Pump Model Revision for RELAX."

8.7 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.

8.8 Siemens Power Corporation, "Generic Mechanical Design Criteria for BWR Fuel Designs," ANF-89-98(P)(A), May 1995.

8.9 Robert E. Moody (NRC) letter to Randall K. Edington (Entergy), "River Bend Station, Unit 1- Issuance of Amendment Re: A Change to the Minimum Critical Power Ratio Safety Limit (SLMCPR) and Changes to the References for the Analytical Methods Used to Determine the Core Operating Limits (TAC No. MB2044)," dated October 3, 2001.

8.10 Siemens Power Corporation, AN-913 (P) (A), Volume I, "CONTRANSA2: A Computer Program for BWRs."

8.11 GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," August 17, 1995.

8.12 GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," September 30, 1996.

8.13 GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," June 28, 1989.

8.14 GL 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," May 2, 1989.

8.15 General Electric, "Generic Evaluation of Boiling Water Reactor Power Uprate," NEDC-31984P, Volume I, July 1991.

8.16 Exxon Nuclear Company, "HUXY: A Generalized Multirod Heatup Code with 10 CFR Part 50, Appendix K, "Heatup Option Users Manual," XN-CC-33 (A).

8.17 Stuart A. Richards (NRC) letter to Michael A. Krupa (Entergy), Safety Evaluation dated December 20, 2001, "Waterford Steam Electric Station, Unit 3; River Bend Station; and Grand Gulf Nuclear Station - Review of Caldon, Inc. Engineering Report ER-157P, "Supplement to Topical Report ER-80P: Basis for Power Uprate with LEFM√™ or LEFM CheckPlus™ System.""

8.18 IEEE 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Generating Stations," February 28, 1974.

8.19 General Electric, "General Electric Thermal Analysis Basis, NEDO-10958-A," November 1973.

8.20 NUREG/CR-3659, "A Mathematical Model for Assessing the Uncertainties of Instrumentation Measurements for Power and Flow of PWR Reactors," February 1985.

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