October 10, 2002

Mr. William A. Eaton
Vice President, Operations GGNS
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION, ISSUANCE OF AMENDMENT RE: 1.7% INCREASE IN LICENSED POWER LEVEL (TAC NO. MB3972)

Dear Mr. Eaton:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 156 to Facility Operating License (FOL) No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications and FOL in response to your application dated January 31, 2002, as supplemented by letters dated June 12, June 25, July 22, September 16, and October 2, 2002.

This amendment increases the licensed power level by approximately 1.7% from 3,833 megawatts thermal (MWt) to 3,898 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,

/RA/

David H. Jaffe, Sr. Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures:
1. Amendment No. 156 to NPF-29
2. Safety Evaluation

cc w/encls: See next page

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<u>DISTRIBUTION</u>:

PUBLIC
PDIV-1 Reading
RidsNrrDlpmPdiv (CHolden)
RidsNrrDlpmPdivLpdiv1 (RGramm)
RidsNrrPMDJaffe
RidsNrrLADJohnson
RidsOgcRp
RidsAcrsAcnwMailCenter
GHill (2)
RDennig, DRIP/RORP (RLD)
RidsRgn4MailCenter (KBrockman)
ZAbdullahi
GThomas
GGeorgiev
CWu
TAlexion
BVaidya
JLee
IAhmed
NTrehan
ASmith
LHurley, RIV

* SE input provided - no major changes made.

Accession No.: ML022630304

**See previous concurrence

OFFICE	PDIV-1/PM	PDIV-1/LA	EMCB/SC	EMCB/SC	SPLB/SC	SPSB/SC(A)	SRXB/SC	EMEB/SC
NAME	DJaffe	DJohnson	LLund*	SCoffin*	SWeerakkody*	FMReinhart*	RCaruso*	KManoly*
DATE	10/3/02	10/3/02	06/27/02	07/11/02	06/19/02	07/11/02	07/24/02	07/11/02

OFFICE	EEIB/SC	EEIB/SC	IEHB/SC	OGC	PDVI-1/SC	PDVI/D	DLPM/D
NAME	JLazevnick*	EMarinos*	CGoodman*	RWeisman**	RGramm	WRuland	JZwolinski
DATE	05/31/02	06/21/02	07/24/02	09/30/02	10/3/02	10/7/02	10/9/02

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ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

ENTERGY MISSISSIPPI, INC.

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 156
License No. NPF-29

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

   A. The application for amendment filed by Entergy Operations, Inc. (the licensee) dated January 31, 2002, as supplemented by letters dated June 12, June 25, July 22, September 16, and October 2, 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, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Operating License and the Technical Specifications, as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-29 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 156, are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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: October 10, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 156

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

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 4	page 4
TECHNICAL SPECIFICATIONS	
1.0-5	1.0-5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 156 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC., ET AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By application dated January 31, 2002 (Reference 8.1), as supplemented by letters dated June 12, June 25, July 22, September 16, and October 2, 2002, (References 8.1a through 8.4, respectively), Entergy Operations Inc., et al. (EOI, Entergy, or the licensee) submitted a request for changes to the Grand Gulf Nuclear Station, Unit 1 (GGNS), Facility Operating License (FOL) and Technical Specifications (TSs). This proposed amendment would increase the licensed power level by approximately 1.7% from 3,833 megawatts thermal (MWt) to 3,898 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 June 12, June 25, July 22, September 16, and October 2, 2002, provided clarifying information that did not change the scope of the original *Federal Register* notice (67 FR 15622, published April 2, 2002) or the original no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

On June 1, 2000, a revision to 10 CFR Part 50, Appendix K, “ECCS [Emergency Core Cooling System] Evaluation Models,” 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 of 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. The original regulation did not require the power measurement uncertainty to be demonstrated, but rather mandated a 2% 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. Reference 8.1 includes a justification of the reduced power measurement uncertainty and the basis for the modified ECCS analysis.

GGNS was originally, and is currently, licensed to operate at a maximum power level of 3,833 MWt, to which a 2% 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. Appendix K has since been revised to permit 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.

Attachment 2 (NEDC-33048P) of Reference 8.1 contained the plant-specific evaluation for the proposed 1.7% power uprate. The licensee subsequently submitted Reference 8.1a, providing a revised version of Attachment 2 to Reference 8.1. By supplemental letter dated October 2, 2002, (Reference 8.1b), the licensee superceded Reference 8.1a in its entirety. The licensee subsequently withdrew Attachment 2 of Reference 8.1. The licensee indicated that the evaluation follows the scope and content of General Electric (GE) licensing topical report NEDC-32938P (Proprietary), Thermal Power Optimization (TPO) licensing topical report (TLTR) (Reference 8.5), for up to 1.5% power uprate. Since Reference 8.5 is based on the generic guidelines and evaluations in the GE licensing topical reports ELTR1 and ELTR2 (References 8.6 and 8.7, respectively), which were reviewed and approved by U.S. Nuclear Regulatory Commission (NRC) staff for extended power uprates in GE boiling water reactors (BWRs) of up to 120% of the original licensed thermal power, it can be reasonably referenced for the proposed 1.7% power uprate at GGNS.

GGNS has installed the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ (√+™) System for FW flow measurement. Use of the LEFM√+™ System will reduce the calorimetric core power measurement uncertainty to < 0.3%. 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% 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).

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 8.8) and December 2001 (Reference 8.9), respectively.

The plant-specific basis for the proposed uprate is provided in the applicable sections of the GE Nuclear Energy topical report included in Reference 8.1b and in Entergy's responses to staff requests for additional information (RAIs) (References 8.2, 8.3, and 8.4). The licensee also stated in Reference 8.2 that the plant TS safety system instrumentation nominal trip setpoints and allowable values do not require any revision due to the power uprate.

2.1 Applicable Regulatory Requirements/Criteria

The staff finds that the licensee, in Sections 3.0, 4.0, and 5.0 of Attachment 1 of Reference 8.1, identified the applicable regulatory requirements. The review and the basis for staff acceptance included 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 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

The licensee submitted safety analysis report (SAR) NEDC-33048P (Attachment 1 to Reference 8.1b) 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 nuclear steam supply system (NSSS) and its components. The power-dependent safety analyses, which are based on 102% of the current reactor thermal power, will remain applicable and bounding at the uprated condition; however, 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 position to reduce the main steam (MS) line flow resistance.

Attachment 1 to Reference 8.1b follows the generic format and content of Reference 8.5. This report is under staff review and is intended to be used for reference in future plant-specific TPO requests. Reference is made to the TLTR in several sections of the GGNS plant-specific TPO report, even though the TLTR report covers power uprates of only up to 1.5%. In response to a staff question regarding the applicability of the TLTR to the 1.7% power uprate, EOI stated that every reference made to the TLTR in the GGNS TPO SAR is valid. The methodology for the analysis of the GGNS TPO addressed the following three approaches: (1) the existing analysis conducted at 102% or greater of current licensed thermal power (CLTP) is bounding for the TPO power uprate; (2) new plant-specific analysis was conducted; or (3) the generic analysis presented in the TLTR is applicable. A confirmation was made that the generic analysis at the 1.5% uprate was valid for GGNS's 1.7% uprate.

The staff finds it acceptable for EOI to refer to Reference 8.5 and the staff believes that this justification is acceptable for the GGNS TPO as discussed below.

The Cycle 13 GGNS core utilizes a mixed core of 800 fuel assemblies, which consist of 240 fresh Framatome (formerly known as Siemens Power Corporation) ATRIUM-10 assemblies, 204 once-burned ATRIUM-10 assemblies, 228 twice-burned GE 11 assemblies, and 128 thrice-burned GE 11 assemblies. The GGNS reload analysis is based on the NRC-approved GE methodology described in NEDE-24011-P-A (Reference 8.10) and Framatome methodologies described among other things, in ANF-91-048(P)(A) (Reference 8.11). The complete list is provided in GGNS TS 5.6.5. The NRC-approved codes and methodologies used for the licensing safety analyses are also referred to in Section 5.0 of the GGNS 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 Final SAR (FSAR). Limiting transient and accident analyses are generally defined as analyses of events that could 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 listed in GGNS TS 5.6.5.

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 assemblies are designed to ensure that (1) they are not damaged during normal steady state operation and AOOs, (2) any damage would not be so severe as to prevent control rod insertion when required, (3) the number of fuel rod failures during accidents 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 (SRP), 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.

Since the uprated core for GGNS will consist of GE 11 and Framatome ATRIUM-10 fuel assemblies, the fuel design criteria are based on the NRC-approved methodology described in ANF-89-98(P)(A) (Reference 8.12) and Reference 8.10. A new mechanical fuel design is not needed to achieve the 1.7% power uprate, even 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 any new fuel designs that do not comply with the NRC-approved fuel design criteria given in Reference 8.10 and Reference 8.12 will require NRC review and approval.

The slightly higher operating power and the increased steam void content will affect the core and fuel performance. Moreover, the licensee may change the power distribution in the reload design to achieve 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 and 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.

Upon introduction of any new fuel type, numerous evaluations are performed as part of the reload process. These evaluations not only confirm that the approved burnup limits are not exceeded, but also address all other impacts that 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 the methods and processes described in the NRC-approved fuel vendor topical reports to perform these analyses and evaluations.

3.1.2 Thermal Limits Assessment

GDC 10 of 10 CFR Part 50, Appendix A, requires that the reactor core and the associated control and instrumentation systems be designed with an 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 limits 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% 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% 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) throughout the cycle using NRC-approved methodologies. In FOL Amendment Number 146 (Reference 8.13), dated April 26, 2001, the staff approved the minimum critical power ratio (MCPR) safety limit for the current fuel cycle, Cycle 12, operation, which included TPO conditions. The staff concluded that the licensee has performed its SLMCPR calculation using the NRC-approved methodology.

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 licensee analyzed the power-dependent transients at 101.7% of the current thermal power as part of the Cycle 13 reload analysis. The change in the MCPR is combined with the SLMCPR to establish the OLMCPR, which ensures that 99% of the rods will not reach boiling transition in the event of an anticipated transient. The licensee has established the OLMCPR at the uprated condition for GGNS.

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 cycle in the reload analysis for future cycles, and these limits will be maintained during operation.

The maximum average 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 has calculated the OLMCPR, the SLMCPR, the LHGR, and the MAPLGHR for the uprated conditions as part of the Cycle 13 reload analysis using NRC-approved methodologies, which is acceptable. It is expected that the licensee will propose appropriate changes to the limits in the TSs and/or the core operating limit report.

3.1.3 Reactivity Characteristics

The GGNS Cycle 13 core has been designed for a TPO uprate to ensure sufficient excess reactivity and that the thermal margin is available. The shutdown margin calculations for the Cycle 13 core have considered the TPO condition. The shutdown margin is 1.04% delta k/k, which is well above the 0.38% value required by TS 3.1.1 and is, therefore, acceptable.

3.1.4 Stability

The licensee stated that the power uprate will not increase the licensed maximum core flow, but the associated control and protective systems, which are based on percent power and percent flow, will be rescaled to the uprated thermal power. GGNS is licensed to operate with a maximum core flow of 105% of rated flow or 118.125 Mlb/hr.

The absolute value of the maximum extended load limit analyses rod line is not changed for the TPO uprate. GGNS utilizes reactor stability Enhanced Option 1-A (E1A). E1A requires immediate protective action when entering the exclusion region. This action can be either a scram or a select rod insert.

The reload validation matrix (RVM) set of stability analyses was performed, and demonstrated acceptable stability performance in the TPO operating condition. The RVM is a set of fully prescribed analyses intended to challenge the stability characteristics of a specific core design. This set of analyses is implemented with each reload, based upon the need for such analyses per reload review criteria established in staff-approved GE topical report NEDO-32339(A) (Reference 8.14). The RVM includes several analytical cases designed to simulate the most severe operational challenges to reactor core stability, based on operating experiences and analytically predicted stability behavior. The RVM includes steady-state conditions, evaluations of startup conditions concerning recirculation pump upshift, evaluations of flow runback events, and evaluations of loss-of-feedwater (LOFW) heating events. This process is described in staff-approved GE topical reports NEDO-31960-A (Reference 8.15) and Reference 8.14. Since the staff has approved this methodology for stability detection and mitigation, the licensee’s use of the methodology is acceptable.

3.1.5 Reactivity Control

3.1.5.1 Control Rod Drives and Control Rod Drive Hydraulic System

The generic discussions in Reference 8.5, Section 5.6.3, and Appendix J of Section 2.3.3 of the same report apply to GGNS. 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. EOI 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) The proposed power uprate may minimally affect the scram timing during transient overpressure conditions. The staff safety evaluation report for Reference 8.7 states that "the plant specific submittal for BWR/6 plants must provide assurance that the scram insertion speeds used in the transient analyses are slower than the requirements in the plant TSs." The licensee confirmed that the GGNS transient analyses apply scram speeds that are slower than the requirements in the TSs.

(3) There must be a minimum pressure differential of 250 psid 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 (DBA). The licensee indicated that the reactor vessel operating and design pressure and temperature values that are used in the existing DBA remain bounding for the proposed 1.7% power uprate. The licensee concluded that the existing GGNS design basis for stresses and fatigue cumulative usage factors of the CRDMs are not affected by the proposed 1.7% power uprate condition. On the basis of its review, the staff concludes that the CRDM will continue to meet its design basis and performance requirements for the proposed 1.7% power uprate conditions.

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% power uprate can be regulated adequately by adjusting the turbine control valve 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 reactor heat balance parameters for the rated and the proposed uprated conditions. The table shows that, for a core flow of 112.5%, the steam flow rate increases by 2% for the uprated conditions. Considering that the steam flow will increase by 2%, that the SRVs will actuate at the current setpoints, and that the current American Society of Mechanical Engineers (ASME) overpressure protection analysis is based on operation at 102% 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.

The ASME Boiler and Pressure Vessel Code (Code) allowable peak pressure for the reactor vessel is 1,375 psig (110% of the design pressure of 1,250 psig), which is the acceptance limit for pressurization events. The licensee analyzed AOOs that may result in the largest overpressure transient on a cycle-specific basis, taking into account the power uncertainty. The most limiting overpressure transient event for GGNS is the MS isolation valves closure (MSIVC) event with high neutron flux scram. The licensee has analyzed the MSIVC at 102% of TPO RTP with the NRC-approved COTRANSA2 methodology in ANF-913(P)(A) (Reference 8.16). The number of SRVs assumed out of service in the analysis is consistent with the number specified in the TS 3.4.4, "Safety/Relief Valves (S/RVs)". The licensee determined that the peak pressure remains below the ASME Code limit of 1,375 psig. Since the licensee used a staff-approved methodology with appropriate analysis input values, the peak pressure is acceptable.

3.2.2 Reactor Pressure Vessel and Internals

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

The licensee indicated that the effect of the proposed uprate for the reactor vessel components except the FW nozzles was evaluated in accordance with the ASME Code, 1971 edition with addenda to and including winter 1972, which is the code of record. In Reference 8.2, the licensee indicated that the FW nozzle, which previously had a modification to its safe end, was evaluated using the 1980 edition of the ASME Code, which was the code of record at the time of its modification. The proposed power uprate does not change the operating reactor pressure and temperature from the current operating condition. There is no change in fuel lift and seismic loads due to the uprate. The current design basis transients remain valid for the proposed power uprate. The LOCA loads were analyzed at 102% power level, which bounds the proposed power uprate conditions. Also, the recirculation design flow does not change for the power uprate. The AP and JR are not affected by the proposed power uprate. The licensee concluded that the current design basis stress and cumulative usage factor (CUF) analyses for the reactor vessel components will continue to meet the Code limits and are, therefore, acceptable for the proposed power uprate.

The licensee evaluated the effect on the reactor internals of the slight increase in the FW flow, temperature, and the RIPDs. The calculated stresses for the affected limiting reactor internals are shown in Table 3-2 of Attachment 1 of Reference 8.1b. The table shows that the calculated stresses for the 1.7% power uprate remain below the Code-allowable limits. As a result of its evaluation, the licensee concluded that the design basis stresses and fatigue usage factors for the reactor internal components will remain unchanged for the proposed 1.7% power uprate. Based on its review of information provided by the licensee, the staff finds the licensee's conclusion acceptable.

The licensee assessed the flow-induced vibration for the proposed power uprate for limiting reactor internal components. The licensee indicated that there is a slight increase in flow-induced vibrations for the shroud, shroud head and separator, steam dryer, fuel channels, and FW sparger because of an approximately 2% increase in steam and FW flow due to the power uprate. Other internal components are not affected, since the maximum core flow and the maximum recirculation drive flow remain unchanged following the proposed 1.7% power uprate. As a result of its evaluation, the licensee concluded that vibration of safety-related internal components due to flow-induced vibration loads will remain within the GE acceptable stress limits of 10 ksi. The staff accepts the licensee's conclusion that the reactor internals will remain adequate and acceptable for the proposed 1.7% power uprate, considering that the acceptable limit of 10 ksi is more conservative then the ASME allowable limit of 13.6 ksi for service cycles equal to 1.0E11.

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

3.2.3 Reactor Vessel Fracture Toughness

The licensee evaluated the fracture toughness of the reactor pressure vessel (RPV) using NRC-approved fracture toughness evaluation procedures. The end-of-life (EOL) fluence is calculated for the TPO uprate conditions, using the fluence for current conditions to evaluate the vessel against the requirements of 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." The results of the licensee's evaluations indicate that:

- The upper shelf energy (USE) remains greater than 50 ft-lb for the design life of the vessel and maintains the margin requirements of 10 CFR Part 50, Appendix G. The minimum EOL USE for beltline materials is 80 ft-lb.

- The beltline material reference temperature of the nil-ductility transition (RT_{NDT}) remains below 200 $^\circ$F.

- The surface fluence increases for EOL (35 effective full-power years (EFPYs)) due to TPO uprate. However, because the current pressure temperature (P-T) curves for

GGNS are based on a vessel thickness of 6.19 inches (without cladding), which has been subsequently updated to 6.44 inches (without cladding) by the TPO evaluation, the net effect for the 1/4T fluence (32 EFPYs) is negligible for TPO. Because the 1/4T fluence contributes to the resulting adjusted reference temperature (ART), there is no change to the ART or shift for EFPYs up to and including 32 EFPYs. In the current GGNS license, the P-T curves for 16, 20, 24, 28, and 32 EFPYs account for Shift values of 64 °F, 71 °F, 77 °F, 83 °F, and 88 °F, respectively. The shift values calculated for TPO are unchanged up to 32 EFPYs. Therefore, the current 16, 20, 24, 28, and 32 EFPY P-T curves are valid with TPO uprate. Due to an increased capacity factor, the EOL EFPY is 35. Thus, prior to operation beyond 32 EFPYs, the P-T curves would be revised to account for a shift value of 91 °F (a 3 °F increase), which represents the shift in RT_{NDT} at 35 EFPYs.

- The 35 EFPY shift is slightly increased and, consequently, requires a change in the ART, which is the initial RT_{NDT} plus the shift. These values are provided in Table 3-1 of Attachment 1 to Reference 8.1b.

- The RPV material surveillance program involves three capsules. The three capsules have been in the reactor vessel since plant startup. One of these capsules is scheduled to be removed after 24 EFPYs of operation, the removal schedule for the second capsule is to be determined, and the third capsule is classified as "Standby." The licensee concluded that the TPO uprate does not require a change to the existing surveillance schedule.

Since the maximum operating dome pressure for the TPO uprate is unchanged from current operation, no change in the hydrostatic and leakage test pressures is required. The licensee concluded that the vessel is still in compliance with the regulatory requirements at TPO uprate conditions.

The staff reviewed the detailed proprietary information provided in Table 3-1 of Attachment 1 to Reference 8.1b, and found that the information provided in Table 3-1 included adequate detailed information to support the licensee's conclusions described above. The staff also reviewed the licensee's analysis and reasoning that the 1.7% power uprate will not have a measurable effect on the RPV fracture toughness and that the RPV will remain in compliance with the requirements stated in 10 CFR Part 50, Appendix G, and found the GGNS RPV fracture toughness will remain in compliance with 10 CFR Part 50, Appendix G, after the 1.7% power uprate is implemented at GGNS. Therefore, the staff concludes that the 1.7% power uprate will not have a measurable effect on the RPV fracture toughness.

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% 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 8.1b. The piping systems evaluated by the licensee included the 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, high-pressure core spray, 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 pressures and temperatures, nor are there any changes in the MS operating and design pressures and temperatures. There is a slight increase in the MS and FW flow rate and in the FW system operating pressure and temperature. The licensee reviewed the design basis calculations associated with the RCPB piping and its support components. The licensee evaluated the MS and attached piping systems, and determined that there are sufficient margins between the calculated stresses (or CUFs) and the allowable limits to accommodate the slight increase (about 2%) in steam flow for the proposed power uprate condition. The licensee also indicated that the increased pressure, temperature, and flow rate in the FW line and its attached piping systems, due to the proposed power uprate, are bounded by the current licensing basis conditions. Therefore, the licensee concluded that the existing design basis of the RCPB piping and supports is adequate and acceptable for operation at the 1.7% uprate conditions. 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% 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. The BOP piping systems that are affected were determined from the uprated reactor and BOP heat balances. The systems affected by the proposed power uprate are MS, extraction steam, turbine bypass, condensate, and FW lines.

The licensee reviewed the piping stress analyses of record. The input parameters (temperature and pressure) used in the design basis piping stress analyses remain 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 GGNS BOP piping and related support systems remain within allowable stress limits in accordance with ASME Code, Section III, 1974 edition through the summer 1975 addenda, and American National Standards Institute (ANSI) B31.1, 1973 edition through summer 1973 addenda, as appropriate. The staff finds acceptable the licensee's conclusion that the BOP systems will operate at the proposed 1.7% power uprate conditions without adverse effects on the piping system and its supports.

As indicated by the licensee in its amendment request, there is no change in the nominal RPV dome pressure. Also, the existing high-energy line break (HELB) analyses were performed assuming 102% of the current power level, which bounds the proposed 1.7% 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% power uprate condition. Based on its review, 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 GGNS was performed based on 102% of the current power level and using a maximum expected differential pressure that bounds the proposed 1.7% power uprate condition. The licensee evaluated its commitments relating to Generic Letter (GL) 95-07 (Reference 8.17) and found that the existing analysis conditions remain bounding for the 1.7% power uprate. The licensee also evaluated its response relating to the GL 96-06 (Reference 8.18) program regarding the overpressurization of isolated piping segments and concluded that the existing evaluation for Reference 8.18 was performed at 102% power and is, therefore, bounding for the proposed power uprate. On the basis of the above review, the staff finds acceptable the licensee's conclusion that the power uprate will have no adverse effects on the safety-related valves, and finds that the licensee's conclusions from the Reference 8.17, Reference 8.18, and GL 89-10 (Reference 8.19) programs regarding safety-related MOV testing and surveillance programs remain acceptable.

3.2.4.3 Flow-Accelerated Corrosion in Piping

The licensee stated that GGNS has established a program for monitoring pipe wall thinning in single- and two-phase high-energy carbon steel piping. This erosion/corrosion program identifies the piping components and locations that should be monitored for flow-accelerated corrosion (FAC). The FAC program considers the guidance of GL 89-08 (Reference 8.20) and defines the criteria for inspecting pipes and components subject to FAC. The GGNS FAC program uses CHECKWORKS software to predict the susceptibility of piping to FAC and to establish a recommended inspection schedule. In addition, the program provides for the expansion of the inspection scope as needed.

The effects of FAC are influenced by fluid velocity, temperature, and moisture content. The licensee stated that the small changes associated with the power uprate will have no adverse effect on piping systems susceptible to FAC and will result in a negligible change to wear rates. The licensee stated that, prior to Refueling Outage 13 (RF13), GGNS will perform a study at the uprated conditions to quantify the impact of the uprate on wear rates. If necessary, the CHECKWORKS model will be updated.

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

3.2.5 Reactor Recirculation System

The reactor recirculation system evaluation described in Section 5.6.2 of Reference 8.5 applies to GGNS.

The power uprate will be accomplished by operating along extensions of the rod and core flow lines on the power/flow map. GGNS is currently licensed to operate at up to a maximum core flow of 105% of the rated flow or 118.125 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% 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 cavitation protection interlock will remain the same in relation to absolute power, since the interlock is based on the FW flow. The licensee pointed out that the ratio of core thermal power level to FW flow remains unchanged during operation at the RTP; therefore, the cavitation interlock remains unchanged.

The staff concludes that the changes associated with the 1.7% 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% power uprate and that all safety and operational aspects of the MSIVs are within previous evaluations. Regarding the main steamline flow restrictors, the licensee states that the requirements remain unchanged for the power uprate because no change in steam break flow occurs (since the operating pressure is unchanged), and that all safety and operational aspects of the flow restrictors are within previous evaluations.

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 plant operations at the 1.7% power uprate condition will have an insignificant impact on the ability of the MSIVs and main steamline 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 8.5 is applicable to GGNS.

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 psig to the maximum pressure corresponding to the lowest opening setpoint for the SRVs. In particular, the LOFW flow transient assumes that the RCIC system will maintain sufficient water level inside the core shroud 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 would 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 only slightly more severe at a higher initial power since there is slightly more stored energy and decay heat to be dissipated and the water level drops faster. The LOFW analysis described in NEDC-31984P (Reference 8.21) is applicable to GGNS.

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

3.2.8 Residual Heat Removal System

The generic discussion provided in Section 5.6.4 of Reference 8.5, and Appendix J, Sections 2.3.1 and 2.3.13 of the same report, is applicable to GGNS.

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel, and remove sensible and decay heat from the primary system and containment following reactor shutdown for both normal and post-accident conditions. The licensee has evaluated the RHR system for various modes of operation, including Low-Pressure Coolant Injection (LPCI) mode, the shutdown cooling mode, the suppression pool cooling mode, the containment spray cooling mode, and the fuel pool cooling assist mode, and found that the RHR system is able to perform its required safety functions based on previous analyses done at greater than or equal to 102% of the core licensed thermal power.

Based on the NRC staff review and 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% 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 the power uprate conditions will not significantly affect the remaining parameters. In addition, the licensee stated that the current capacity of the RWCU system will be sufficient and can be adjusted to accommodate the power uprate.

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 it will not significantly affect the water chemistry performance of the reactor and, therefore, will not significantly affect the performance requirements of the RWCU system.

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. Reference 8.1 states that the previous containment evaluations are bounding for the 1.7% power uprate because they were performed at greater than or equal to 102% of the current licensed thermal power. 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 pressure and temperature response of the containment, the long term temperature response of the suppression pool, the containment dynamic loads, and containment isolation.

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 the containment system performance will not be affected by the 1.7% power uprate.

3.3.2 Emergency Core Cooling Systems

The ECCS for GGNS includes the high-pressure core spray (HPCS) system, the LPCI mode of the RHR system, the low-pressure core spray (LPCS) system, and the automatic depressurization system (ADS).

3.3.2.1 High-Pressure Core Spray System

The HPCS system (operating with other ECCS systems) is designed to maintain reactor water inventory during a LOCA,. 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 during isolation transients with LOFW. The HPCS system is designed to operate from normal offsite auxiliary power or from its dedicated emergency diesel generator.

The HPCS system is required to start and operate reliably over its design operating range. During the LOFW event and isolation transients with LOFW, the RCIC system maintains water level above the top of active fuel (TAF).

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 peak cladding temperature (PCT) following a LOCA and ensure core coverage above 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 8.5. The maximum reactor pressure at which the HPCS system must be capable of injecting into the vessel for the RCIC system backup function was selected based on the upper analytical values for the second lowest group of SRVs operating in the low-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 the 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 analysis acceptable.

3.3.2.2 Low-Pressure Core Spray System

The LPCS system initiates automatically in the event of a LOCA. In conjunction with other ECCS systems, the LPCS system provides adequate core cooling for all LOCA events. The system also provides spray cooling for long-term core cooling in the event of 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 8.5 is applicable to GGNS.

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

3.3.2.3 Low-Pressure Coolant Injection Mode

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 further 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 mode capability (see Section 3.3.3 below) and demonstrates that the system provides adequate core cooling, the staff finds the analysis acceptable.

3.3.2.4 Automatic Depressurization System

The ADS uses the SRVs to reduce reactor pressure after a small-break LOCA with HPCS system failure, allowing the LPCI and LPCS systems 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 (Reference 8.11), EXEM (Reference 8.11), and HUXY (Reference 8.22) models, while the GE fuel was analyzed with GE's NRC-approved SAFER/GESTR model (Reference 8.10). These analyses were performed at 4,105 MWt (102% of 105% of current licensed power level) for the 3,898 MWt 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 1850 °F, peak metal water reactions less than 3%, and core-wide metal-water reactions less than 0.1%. These results comply with the 10 CFR 50.46 requirements of less than 2200 °F PCT, less than 17% cladding oxidation, and less than 1% core-wide-metal water reaction.

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 condition. Also, because previous containment analyses were based on 104.2% of the current licensed thermal power, there is no change in the available NPSH for systems using suppression pool water. Therefore, the 1.7% power uprate will not affect performance of the ECCS pumps.

The existing DBA assumes an accident occurs at 102% of 105% of the current licensed thermal power. Therefore, the existing analysis bounds the 1.7% power uprate. Also, since the LOCA analysis is based on NRC-approved methodology and codes, and the assumed power is bounding, the staff finds acceptable the licensee's assessment that the ECCS will perform as designed and analyzed at the uprated conditions.

The staff finds EOI's ECCS performance evaluation acceptable because the analytical models and codes are based on the NRC-approved methodology described in Reference 8.11 and Reference 8.10, 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 design basis accident. Habitability (including control room operator doses) following a postulated accident from the 1.7% power uprate condition is unchanged because the main control room atmospheric control system had previously been evaluated for accident conditions from 102% of current licensed thermal power. This evaluation is bounding for the proposed 1.7% power uprate.

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 the licensee's existing analysis for the main control room atmospheric control system is bounding for the 1.7% power uprate.

3.3.5 Standby Gas Treatment System and Main Steam Isolation Valve 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% of the current licensed thermal power.

The MSIV leakage control system directs MSIV leakage flow to the secondary containment, where radioactive material is processed by the SGTS. The MSIV leakage control system is adequate for power uprate, since the current evaluations have been performed at 102% of the current licensed thermal power.

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 the licensee's existing analysis for the SGTS and the MSIV leakage control system remains valid for the 1.7% 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% power uprate, and the hydrogen production due to radiolytic decomposition is unchanged because the system was previously evaluated for accident conditions at 104.2% of the current licensed thermal power.

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 the licensee's existing analysis bounds the 1.7% power uprate, and the impact on the hydrogen recombiners is negligible.

3.4 Instrumentation and Controls

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant NSSS. This calculation is called a "heat balance" for a BWR. The accuracy of this calculation depends primarily upon the accuracy of FW flow and FW net enthalpy measurements. Thus, an accurate measurement of FW flow and temperature will result in an accurate calorimetric calculation and an accurate calibration of the nuclear instrumentation.

The instrumentation for measuring the 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. Typically, the FW temperature is measured using resistance temperature detectors mounted in the pipe. 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. This causes the licensee to generate proportionately less electrical power when the plant is operated at its indicated thermal power rating.

The use of the transit time methodology with ultrasonic pulse transmission in multiple acoustic paths across pipe cross sections, as utilized by the Caldon LEFM√+™ System technology, improves the accuracy of the measurement of FW flow and reduces the uncertainty of the flow measurement.

Entergy stated that the Caldon LEFM√+™ System is designed and manufactured in accordance with the Caldon 10 CFR Part 50, Appendix B, quality assurance program, and the system software and laboratory calibration tests are required to meet 10 CFR Part 50, Appendix B, requirements. The system software was developed under the Caldon Verification and Validation (V&V) program, which meets the criteria of ANSI/Institute of Electrical and Electronic Engineers (IEEE) Standard 7-4.3.2, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Generating Stations," and ASME Standard NQA-2A-1990, "Quality Assurance Requirement for Nuclear Facility Applications." The V&V program is consistent with the guidance of Electric Power Research Institute topical report TR-103291s, "Handbook for Verification and Validation of Digital Systems," and includes requirements for user notification of important deficiencies. All conditions adverse to quality are handled in accordance with the Entergy corrective action program. The licensee also stated that the Caldon LEFM√+™ System software will be controlled under the GGNS software quality assurance program, which provides for appropriate vendor notification and error reporting.

The staff safety evaluation (Reference 8.23) on Caldon topical report ER-80P included four additional requirements to be addressed by licensees who wish to reference the Caldon topical report (1) maintenance and calibration of the Caldon LEFM√+™ System and other instrumentation affecting heat balance, (2) hydraulic configuration of the installed Caldon LEFM√+™ System, (3) processes and contingencies for an inoperable Caldon LEFM√+™ System, and (4) methodology for calculating the Caldon LEFM√+™ System and plant core power measurement uncertainties. Reference 8.2 addressed each of the four requirements along with the following commitments for continuous compliance:

- Calibration and Maintenance work will be performed in accordance with Caldon recommendations.

- The LEFM√+™ System software will be controlled under the GGNS software quality assurance program.

- This requirement (LEFM AOT [allowed outage time]) will be controlled by the GGNS Technical Requirements Manual.

- If the plant experiences a down power of greater than 10% during the 72 hour period, then the permitted maximum power level would be reduced to 3,833 MWt upon return to full power, since a plant transient may result in calibration changes of the venturis (e.g., defouling).

- With an LEFM out of service for more than the above allowed outage time, GGNS will limit power to the original licensed power level of 3,833 MWt.

The staff finds that Entergy's responses 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.

On the basis of References 8.1 through 8.4, the staff finds that the GGNS thermal power measurement uncertainty using the LEFM√+™ System is limited to ±0.3% of the reactor thermal power and, therefore, can support the proposed 1.7% thermal power uprate. The staff also finds that the licensee adequately addressed the four additional requirements outlined in Reference 8.2.

3.5 Electrical Systems

The GGNS distribution system consists of various auxiliary electrical systems to provide electrical power during all modes of operation and shutdown conditions. The electrical distribution system has been previously evaluated to conform to 10 CFR Part 50, Appendix A, GDC-17. The plant has also been previously evaluated for environmental qualification (EQ) of electrical equipment, 10 CFR 50.49, and station blackout (SBO), 10 CFR 50.63. The basis for the NRC staff's power uprate evaluation also included 10 CFR Part 50, Appendix A, GDC-17; 10 CFR 50.49; and 10 CFR 50.63.

The following is the NRC staff's power uprate evaluation of grid stability, main generator, main transformers, emergency diesel generators, SBO, and EQ.

3.5.1 Grid Stability

The licensee performed the grid stability analyses in accordance with the guidance in the NRC SRP, NUREG-0800, Section 8.2.III.1.f. For the dynamic stability study, the analysis tripped the plant and applied faults which lasted up to 15 cycles. Only one case during an off-peak condition went unstable after 14 cycles, which is not considered a problem since it is beyond the time for backup breakers to respond, assuming a failed or stuck breaker in conjunction with the fault occurring (typically, backup breakers trip within 7 cycles). The power uprate will have a negligible impact on how the grid reacts to the main generator tripping, since a 500 kV system has the capacity to account for this additional loss. For the steady-state stability study, the analysis assumed multiple failures over and above Reference 8.24 requirements for the 500 kV line and it met the 0.975-per-unit requirement for minimum voltage.

Based on the above, the staff concludes that the plant continues to meet GDC-17 for grid stability with this power uprate.

3.5.2 Main Generator

The main generator is rated at 1525 megavolt amperes (MVA) and 22 kV with an operating point of 1372.5 MWe at a 0.90 power factor. Since the generator will continue to operate below its design rating of 1525 MVA, the power uprate does not affect the generator auxiliaries listed below:

- 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 power uprate. Therefore, operating the main generator at the uprated power conditions is acceptable.

3.5.3 Main Transformer

The main transformer has a rating of 510 MVA (1 phase)/1530 MVA (3 phase) at a 65 °C forced oil and air (FOA) winding temperature rise. The transformers will operate within applicable limits at power uprate conditions and are, therefore, acceptable.

3.5.4 Isophase Bus

The isophase bus (main transformer delta bus) is designed for 22,000 amperes per phase. The associated power, 539 MVA/phase, is well above the capacity of the transformer or main generator. Therefore, the isolated phase bus duct will continue to support plant operation under uprated conditions.

3.5.5 Service Transformers

Each service transformers is rated at 168 MVA at a 65 °C FOA, which is well above total station load of approximately 84 MVA. The associated cooling equipment will also support power uprate for continuous operation with no modifications.

3.5.6 Onsite Power

Station loads under normal operation are computed based on equipment nameplate data with conservative demand factors applied. The only identifiable change in electrical load demand is associated with condensate, condensate booster, and heater drain pumps. These pumps experience increased flow and pressure due to the TPO uprate conditions. Because these changes are small, the motor demand for each of these loads remains bounded by the existing calculations. Accordingly, there are no changes in the onsite distribution system design basis loads or voltages due to the TPO conditions. Therefore, the large station auxiliary loads and associated cables are considered adequate as installed, and the motors will continue to satisfactorily perform their intended functions.

3.5.7 Emergency Diesel Generators

There is no change to the safety-related loads at uprate conditions and, therefore, the emergency diesel generators will not be affected by the power uprate and can perform their safety-related functions during a loss-of-offsite power/LOCA.

3.5.8 Environmental Qualification of Electrical Equipment

Because the existing specified parameters remain bounding, no review of equipment was necessary. Conservatisms in accordance with IEEE 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Station," published on February 28, 1974, were originally applied to the environmental parameters, and no change is needed for the TPO uprate.

EQ for safety-related electrical equipment located inside the containment is based on the MS line break (MSLB) and/or LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environments expected to exist during normal plant operation. The current accident conditions for temperature and pressure are based on an analysis initiated from ≥ 102% of the current thermal power level. Due to the TPO uprate, normal temperatures may increase slightly near the FW and reactor recirculation lines. These will be evaluated through the EQ temperature-monitoring program, which tracks such information for equipment aging considerations. The current radiation levels under normal plant conditions also increase slightly. The current plant environmental envelope for radiation is not exceeded by the changes resulting from the TPO uprate.

The accident temperature, pressure, and humidity environments used for the qualification of equipment outside containment result from an MSLB in the pipe tunnel or other HELBs. Some of the HELB pressure and temperature profiles increase by a small amount due to the TPO uprate conditions. However, there is adequate margin in the qualification envelopes to accommodate the small changes. Based on these considerations, operating at the uprated power condition is acceptable and in conformance with 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 it will not result in a delay in removing the RHR system from service (i.e., the duration of supplemental cooling will not be increased). The licensee has 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 spent 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 increase will not significantly increase the operational doses to personnel or equipment. 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 NRC staff review 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% uprated power level.

3.6.2 Water Systems

The safety-related standby service water (SSW) system provides cooling to the RHR heat exchangers, diesel generators, and ECCS equipment during and following a design basis accident. The heat loads generated by the diesel generators and the ECCS equipment are not affected by the power uprate. The 1.7% power uprate will increase the heat loads on the RHR heat exchangers and room coolers due to the increase in suppression pool temperature; however, the increased heat loads are acceptable since the containment response analysis was based on a core power level of 104.2% of the current licensed thermal power.

Regarding nonsafety-related heat loads, the plant service water/radial well systems are designed to cool plant auxiliary equipment and provide makeup to the circulating water cooling tower, SSW system, and fire protection system during normal operating and normal shutdown conditions. The 1.7% power uprate will slightly increase makeup flow to the cooling tower and there will be slight increases in heat load from the turbine building closed cooling water system and the component cooling water (CCW) system; however, the increase in system demand and heat loads from these sources is within the design of the plant service water/radial well systems.

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% 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 licensee's evaluation confirms that the condenser, circulating water system, and heat sink are adequate for the power uprate.

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

The power-dependent heat loads on the turbine building closed cooling water system, which increase due to the 1.7% power uprate, are the coolers for the isophase bus, turbine, and generator. The remaining heat loads are not strongly dependent on reactor power and do not increase significantly. The licensee has determined that the turbine building closed cooling water system has sufficient capacity to remove the additional heat load.

The ultimate heat sink (UHS) is provided by the SSW cooling towers and basins. The post-LOCA UHS water temperature will increase slightly due to the 1.7% power uprate, primarily due to higher reactor decay heat. The licensee has determined that the existing UHS system provides a sufficient quantity of water at less than or equal to design temperature following a design basis LOCA to remove the additional heat load.

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 plant operations at the proposed 1.7% 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.

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. Because the uprate will not change the operating parameters of the SLC system, the staff concludes that the SLC system 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, drywell, auxiliary building (including the fuel handling area), and the radwaste building.

The 1.7% power uprate results in a minor increase in heat load caused by the slightly higher FW process temperature (approximately 2 °F). The increased heat load is within the margin of the steam tunnel area coolers. In the drywell, the increased heat load due to the increased 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 due to the increase in the FW process temperatures are less than 2 °F. In the auxiliary 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 NRC staff review 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 detection and suppression is not expected to be impacted by plant operations at the proposed 1.7% 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 do not change, and the operator actions required to mitigate the consequences of a fire are not affected by the uprated conditions.

The GGNS Appendix R fire event analysis assumes an operating power level of 104.2% of the current licensed thermal power at the start of the fire event, which bounds the 1.7% power uprate conditions.

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 the safe shutdown systems and procedures used to mitigate the consequences of a fire will continue to meet 10 CFR 50.48 and 10 CFR Part 50, Appendix R, and will not be affected by plant operations at the proposed 1.7% power uprate.

3.7 Power Conversion Systems

The GGNS power conversion systems and their support systems (including the turbine generator, condenser and steam jet air ejectors, turbine steam bypass, and the FW and condensate systems) are designed for 105% of the current licensed thermal power rated steam flow. Reference 8.1 states that the proposed 1.7% power uprate will increase the rated steam and FW flows by 2%, which is bounded by the 105% design value.

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. The steam flow to the turbine due to the power uprate will increase from 15.8 Mlb/hr to 16.0 Mlb/hr, which is within the turbine's design flow of 16.3 Mlb/hr based on a valves-wide-open reactor steam flow. In addition, the rotor missile and turbine overspeed analyses were determined by the licensee to have adequate margin to bound the 1.7% power uprate condition.

The design margin in the condenser heat removal capability can accommodate the additional heat rejected for operation at the uprated power condition. Operational conditions such as cleanliness, tube plugging, and circulating water temperature cause more significant variations in the condenser back pressure than the small additional heat rejection caused by the power uprate. Regarding the steam jet air ejectors, air leakage into the condenser does not increase as a result of the 1.7% 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 was originally designed for a steam flow capacity of at least 35% of the guaranteed reactor steam flow at the current licensed thermal power. The steam bypass capacity at the 1.7% uprated condition remains above 35% of the uprated steam flow. In addition, the steam bypass system is not safety-related and is not credited in the GGNS transient analysis.

The FW and condensate systems are not safety-related; however, their performance may have an effect on plant operation at the 1.7% 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 turbine controls are capable of maintaining water control at the uprated conditions. The condensate demineralizers will experience slightly higher loadings at the uprated condition, which will result in sightly reduced run times; however, the reduced run times are acceptable.

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 the power conversion systems can accommodate plant operations at the proposed 1.7% 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 reactor water cleanup filter demineralizers. The licensee has reviewed the plant operating effluent reports and the slight increases expected from the 1.7% 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% 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 H_2 and O_2), 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 NRC staff review 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% 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, -15, and -20. 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 8.24 provides further guidelines: (1) pressure in the reactor coolant and MS system should be maintained below 110% 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 GGNS updated FSAR (UFSAR) contains the DBAs that evaluate 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 OLMCPR. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

The GGNS operating Cycle 13 reload analyses have been developed considering the TPO power level using Framatome’s and GE’s approved reload methodology as listed in GGNS TS 5.6.5. In Reference 8.13, the staff approved the EOI request to revise the TSs and the associated Bases for the Cycle 12 operation. The staff concluded that the revisions were acceptable, since the analyses for the SLMCPR were based on NRC-approved methodologies.

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

3.9.2 <u>Radiological Analysis of Design Basis Accidents</u>

In addition to the material included in Reference 8.1, the staff reviewed relevant information in the GGNS UFSAR and in License Amendment No. 145 dated March 14, 2001. The staff reviewed the following four design basis accidents for the proposed power uprate:

- LOCA
- fuel handling accident (FHA)
- CRD accident (CRDA)
- MSLB accident (MSLBA)

In the GGNS design, no instrument or sample line connected to the RCPB penetrates the primary containment. Therefore, the failure of small lines carrying primary coolant outside containment is not applicable. The staff finds that, in Section 15.6 of the GGNS UFSAR, the licensee evaluated the radiological consequences of the MSLBA at a power level of 3,993 MWt, which is 104% of the current power rating of 3,833 MWt, bounding 1.7% power uprate. In Reference 8.26, the licensee and staff analyzed the remaining three design basis accidents (LOCA, FHA, and CRDA) at a power level of 3,910 MWt, which is 102% of the current power rating, also bounding 1.7% power uprate. In general, the inventory of fission products in the reactor core and the quantity of radioactive material in the reactor coolant system are directly proportional to the reactor thermal power.

The staff concludes that there is reasonable assurance that GGNS operation at the increased reactor thermal power of 3,898 MWt will not result in postulated design basis accident radiological consequences that exceed the analysis results currently documented in the GGNS UFSAR and Reference 8.26.

3.9.3 <u>Special Events</u>

3.9.3.1 <u>Anticipated Transient Without Scram</u>

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:

- 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

- 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

- equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS

The GGNS meets the ATWS mitigation requirements defined in 10 CFR 50.62 using the following:

- boron injection equivalent to 86 gpm

- an ARI system

- an automatic recirculation 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 remains below 1500 psig), and containment integrity (the containment temperature and pressure must not exceed the design limit).

Appendix L of Reference 8.5 presents a generic evaluation of an ATWS event for a TPO uprate. This evaluation is applicable to GGNS. However, this evaluation is based on the ATWS response of GE fuel assemblies and GGNS is transitioning from GE to Framatome fuel assemblies.

In Reference 8.4, Entergy stated that Framatome has analyzed the limiting ATWS pressurization events, MSIVC, and pressure regulator failure open at the TPO power level and concluded that there is 237 psi margin to the 1500 psig ATWS peak pressure limit. The ATWS analysis was performed using cycle-specific core characteristics and an NRC-approved code (COTRANSA2). To illustrate the effects of the ATRIUM-10 fuel introduction, Framatome performed two ATWS pressurization analyses at the TPO power level with 26% ATRIUM-10 (Cycle 12) and 56% ATRIUM-10 (Cycle 13) in the core. The two batch sizes of ATRIUM-10 fuel resulted in a peak ATWS vessel pressure difference of 2 psig. Therefore, the staff concludes that GGNS has sufficient ATWS peak pressure margin available as the core transitions to the full ATRIUM-10 fuel and operates at the TPO power level.

The discussion of the suppression pool temperature response to an ATWS event in Reference 8.5 is applicable to GGNS. GGNS has sufficient suppression pool temperature margin, based on a GE core containment analysis performed at the current RTP. Framatome evaluated the response of the GGNS ATRIUM-10 core operating at the TPO power level on the suppression pool heatup, stating that the impact of any fuel differences would be negligible when the steam blowdown into the suppression pool is integrated over the duration of the ATWS event analysis. Since the end-of-cycle RPT occurs early in the ATWS containment analysis event, the peak suppression pool temperature is reached while the reactor is operating at the natural recirculation condition. Therefore, the TPO is expected to have a small impact on the amount of energy deposited in the suppression pool, since the pool heatup occurs while at natural recirculation operating conditions. Therefore, the staff concludes that GGNS has sufficient suppression pool temperature margin to accommodate a 1.7% power uprate.

Based on the justification provided in the TLTR and the analyses performed by Framatome, the RAI responses, and the available margin for peak ATWS parameters, the staff concludes that the licensee evaluation is acceptable. Accordingly, the staff concludes that GGNS meets the ATWS rule requirements specified in 10 CFR 50.62.

3.9.3.2 Station Blackout

Appendix L of Reference 8.5 provides a generic evaluation of a potential loss of all alternating current power supplies based on previous plant response and coping capability analyses for typical power uprate projects. The previous power uprate evaluations (for similar BWR plants) have been performed according to the applicable bases for the plant (e.g., the bases, methods, and assumptions of Regulatory Guide 1.155, "Station Blackout," August 1988). This evaluation is for confirmation of continued compliance with 10 CFR 50.63. Applicable operator actions have previously been assumed consistent with the plant emergency procedure guidelines. These are currently accepted procedures for each plant and SBO analysis. For the TPO uprate, there is no significant change in the time available for the operator to perform these assumed actions. Table L-3 of TLTR provides a basis that the effect of the power uprate on SBO is relatively small. The plant currently has margins of 56,000 gallons to the available condensate storage inventory volume and 10 °F to the peak containment temperature limit. Therefore, a plant specific SBO analysis was not required for GGNS. The plant coping capability with an SBO event for the required duration of 4 hours stays unchanged.

The licensee reviewed the SBO plant response and coping evaluations that were originally performed to satisfy the requirements in 10 CFR 50.63. The licensee found that plant response to coping capabilities for an SBO event is affected slightly by the 1.7% power uprate, due to an increase in decay heat. However, there are no changes to the systems or equipment used to respond to an SBO, nor is the required coping time changed. The licensee stated that the plant continues to meet the requirements of 10 CFR 50.63 after the power uprate.

Based on its review, the staff concludes that the uprate does not adversely affect the ability of the plant to mitigate a postulated SBO event for the uprate conditions.

3.10 Other Evaluations

3.10.1 High-Energy Line Break Analyses

The licensee stated that, since the 1.7% 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 UFSAR. At the uprated power level, HELBs outside the drywell would result in an insignificant change in the subcompartment pressure and temperature profiles. The licensee's 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% 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% power uprate. Therefore, the existing pipe whip restraints and jet impingement shields are adequate.

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 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 its change control process requires the identification and update of the affected operating procedures associated with a modification. The procedures that impact plant operation have been identified and will be revised prior to operation above the current licensed thermal power level.

The staff finds the licensee's response acceptable because the licensee has identified the plant procedures that will be affected by the 1.7% 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 power uprate conditions, operator responses to transient, accident, and special events are not affected. Operator actions for maintaining safe shutdown, core cooling, and containment cooling 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 the LEFM was installed during the previous outage and the out-of-service indication for the LEFM will be installed prior to operation above the current licensed thermal power level. The LEFM status information is made available to the operator on computer displays in the control room. Operators will be required to check the LEFM status at least once every 12 hours. This will be administratively controlled by the Technical Requirements Manual (TRM). It is likely that operators would identify that the LEFM is unavailable sooner than the required check (at least once every 12 hours) because routine computer monitoring, which is more frequent than 12 hours, also indicates LEFM status. There is no immediate action to be taken in response to the LEFM being out of service, as the AOT for the LEFM is 72 hours and the LEFM being out of service does not cause a change in core thermal power. Regarding other changes, the licensee stated that minor changes to the power/flow map and flow-referenced setpoint will be communicated through normal operator training.

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 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 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 conditions. Minor changes to the power/flow map, flow-referenced setpoint, and changes to the TSs 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 physical changes are required to the simulator to reflect the power uprate conditions. Simulator software changes and validation are controlled in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.5-1998, "Nuclear Power Plant Simulators for Use in Operator Training and Examination." Simulator software changes reflecting the new full-power initiating conditions have been tested to confirm the simulator fidelity at the uprated conditions. These simulator changes will be implemented for operator training in the next training cycle, which began in July 2002.

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 3,833 MWt to 3,898 MWt:

- Paragraph 2.C.(1) of FOL NPF-29, "Maximum Power Level," would be revised to authorize operation of the facility at reactor core power levels not in excess of 3,898 MWt (100% of rated power).

- The definition of RTP in TS 1.1 would be revised to state that the RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3,898 MWt.

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 :			
The reactor thermal power will be administratively controlled at a level consistent with the accuracy of the available instrumentation until the LEFM CheckPlus™ system is returned to an operable status. The administrative controls will be added to the GGNS technical requirements manual.		X	upon implementation

COMMITMENT	TYPE (Check One) ONE-TIME ACTION	TYPE (Check One) CONTINUING COMPLIANCE	SCHEDULED COMPLETION DATE (If required)
The plant erosion/corrosion program currently monitors the affected systems. Continued monitoring of the systems provides confidence in the integrity of susceptible high-energy piping systems. Appropriate changes to piping inspection frequency will be implemented to ensure adequate margin exists for those systems with changing process conditions. (TSAR Section 3.5.2)		X	upon implementation
PCS (pressure control system) tests will be performed during the power ascension phase (Section 10.4). (TSAR Section 5.2.1)	X		upon implementation
Per the guidelines of Appendix L of the TLTR, 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. (TSAR Section 5.2.2)	X		upon implementation
In preparation for operation at TPO uprated conditions, routine measurements of reactor and system pressures and flows and vibration measurements on selective rotating equipment will be taken near 95% and 100% of CLTP, and retaken at 100% of TPO RTP. (TSAR Section 10.4)	X		upon implementation

	TYPE (Check One)		SCHEDULED COMPLETION DATE (If required)
COMMITMENT	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Demonstration of acceptable fuel thermal margin will be performed prior to power ascension to the TPO RTP at the 100% CLTP steady-state heat balance point. Fuel thermal margin will be calculated for the TPO RTP point after the measurements taken at 95% and 100% of CLTP to project the estimated margin. (TSAR Section 10.4)	X		upon implementation
The response of the pressure and FW level control systems will be recorded at each steady-state point defined above to demonstrate acceptable operational capability. Water level changes of $\pm$3 inches and pressure setpoint changes of 3 psi will be used to evaluate performance. (TSAR Section 10.4)	X		upon implementation
Minor changes to the power/flow map, flow-referenced setpoint, and the like will be communicated through normal operator training. Simulator changes and validation for the TPO uprate will be performed in accordance with ANSI/ANS 3.5-1985. (TSAR Section 10.6)		X	upon implementation
Prior to operation beyond 32 EFPYs, the P-T curves would be revised to account for a shift value of 91F (a 3F increase), which represents 35 EFPY.	X		upon implementation
The following commitments are from Reference 8.2 :			

COMMITMENT	TYPE (Check One)		SCHEDULED COMPLETION DATE (If required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Calibration and maintenance work will be performed in accordance with Caldon recommendations.		X	
The LEFM,√+™ System software will be controlled under the GGNS software quality assurance program.		X	
This requirement (LEFM AOT) will be controlled by the GGNS technical requirements manual.		X	
If the plant experiences a down power of greater than 10% during the 72-hour period, the permitted maximum power level would be reduced to 3,833 MWt upon return to full power, since a plant transient may result in calibration changes of the venturis (e.g., defouling).		X	
With an LEFM out of service for more than the above allowed outage time, GGNS will limit power to the original licensed power level of 3,833 MWt.		X	
The following commitments are from Reference 8.3 :			

COMMITMENT	TYPE (Check One)		SCHEDULED COMPLETION DATE (If required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The components that could be affected by TPO and have low margin and high wear rates will be inspected in the upcoming RF12. Prior to RF13, GGNS will perform a parametric study at the uprated conditions to quantify the impact of TPO on GGNS wear rates and update the CHECWORKS model if necessary.	X		
Minor changes similar to the changes to the technical specifications, power/flow map, and flow-referenced setpoint will be communicated through routine operator training prior to operation at the uprated power level.	X		
The simulator changes reflecting the new full-power initiating conditions will be implemented for operator training in the next training cycle, which begins July 2002.	X		
The out-of-service indication for the LEFM will be installed prior to operation above the CLTP.	X		
The procedures that impact plant operation have been identified and will be revised prior to operation above CLTP.	X		
The software changes for the plant process computer for power uprate will be implemented prior to operation above CLTP.	X		

COMMITMENT	TYPE (Check One)		SCHEDULED COMPLETION DATE (If required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The following commitment is from Reference 8.4 :			
Operators will be required to check the LEFM status at least once every 12 hours. This will be administratively controlled by TRM.		X	Prior to use of the amendment.

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 Mississippi 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 15622, published April 2, 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 GNRO-2002/00008, William A. Eaton (Entergy) letter to NRC, "Appendix K Measurement Uncertainty Recovery - Power Uprate Request," dated January 31, 2002. (Attachment 2 was withdrawn and superceded by Amendment 2, Reference 8.1b)

8.1a GNRO-2002/00085, William A. Eaton (Entergy) letter to NRC, "Safety Analysis Report, Revision 1, Appendix K Measurement Uncertainty Recovery Power Uprate," dated September 16, 2002. (Attachment 1 and 2 superceded by Reference 8.1b)

8.1b GNRO-2002/00087, Jerry C. Roberts (Entergy) letter to NRC, "Safety Analysis Report, Revision 2, Appendix K Measurement Uncertainty Recovery Power Uprate," dated October 2, 2002.

8.2 GNRO-2002-00049, William A. Eaton (Entergy) letter to NRC, "Response to Request for Additional Information, Appendix K Measurement Uncertainty Recovery - Power Uprate Request," dated June 12, 2002.

8.3 GNRO-2002/00054, William A. Eaton (Entergy) letter to NRC, "Response to Request for Additional Information, Part 2, Appendix K Measurement Uncertainty Recovery - Power Uprate Request," dated June 25, 2002.

8.4 GNRO-2002/00063, William A. Eaton (Entergy) letter to NRC, "Supplement to Amendment Request, Appendix K Measurement Uncertainty Recovery - Power Uprate Request," dated July 22, 2002.

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

8.6 GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (ELTR1), Licensing Topical Report NEDC-32424P-A, Class III (Proprietary), February 1999; and NEDC-32424, Class I (Nonproprietary), April 1995.

8.7 GE Nuclear Energy, "Generic Evaluation of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR2), Licensing Topical Report NEDC-32523P-A, Class III (Proprietary), February 2000; NEDC-32523P-A, Supplement 1, Volume 1, February 1999, and Supplement 1, Volume II, April 1999 (Proprietary).

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

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

8.11 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."

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

8.13 S. Patrick Sekerak (NRC) letter to William A. Eaton (Entergy), "Grand Gulf Nuclear Station, Unit 1- Issuance of Amendment Re: Revision of the Minimum Critical Power Ratio Safety Limit for Cycle 12 Operation (TAC No. MB0514)," dated April 26, 2001.

8.14 General Electric Nuclear Energy, Licensing Topical Report, "Reactor Stability Long-Term Solution: Enhanced Option 1-A," NEDO-32339(A), December 1996.

8.15 General Electric Nuclear Energy, "BWR Owners Group Long-Term Stability Solution Licensing Methodology," NEDO-31960-A, and Supplement 1, April 1996.

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

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

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

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

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

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