March 29, 2002

Mr. Joseph E. Venable
Vice President Operations
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: APPENDIX K MARGIN RECOVERY - POWER UPRATE REQUEST (TAC NO. MB2971)

Dear Mr. Venable:

The Commission has issued the enclosed Amendment No. 183 to Facility Operating License (FOL) No. NPF-38 for the Waterford Steam Electric Station, Unit 3 (Waterford 3). The amendment consists of changes to the FOL and Technical Specifications (TSs) in response to your application dated September 21, 2001, as supplemented by letters dated December 10, 2001, January 16, and January 21, 2002.

The amendment changes the Waterford 3 FOL and TS associated with an increase in the license power level. The changes increase the power level from 3,390 Megawatts thermal (MWt) to 3,441 MWt. The increase results from increased feedwater flow measurement instrumentation accuracy.

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/
N. Kalyanam, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures: 1. Amendment No. 183 to NPF-38
2. Safety Evaluation

cc w/encls: See next page

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ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 183
License No. NPF-38

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

   A. The application for amendment by Entergy Operations, Inc. (EOI) dated September 21, 2001, as supplemented by letters dated December 10, 2001, January 16, and January 21, 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 Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2. of Facility Operating License No. NPF-38 is hereby amended to read as follows:

   2. Technical Specifications and Environmental Protection Plan

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 183 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Changes to the Technical Specifications

Date of Issuance: March 29, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 183

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

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

Remove		Insert
	License	
Page 4		Page 4
	Technical Specification	
1-6		1-6

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 183 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated September 21, 2001 (Reference 1), as supplemented by letters dated December 10, 2001 (Reference 2), January 16, 2002 (Reference 3), and January 21, 2002 (Reference 4), Entergy Operations, Inc. (Entergy or the licensee) submitted a request for changes to the Waterford Steam Electric Station, Unit 3 (Waterford 3), Operating License and Technical Specifications (TSs). The requested changes would increase the power level from 3,390 Megawatts thermal (MWt) to 3,441 MWt, an approximately 1.5 percent increase.

The request is based on the installation of a Caldon Leading Edge Flow Meter (LEFM) CheckPlus (LEFM ✓+™) system in the feedwater pipe from the main feedwater header, which reduces the flow and temperature uncertainties, and the revision of Appendix K to Title 10, *Code of Federal Regulations*, Part 50 (10 CFR Part 50), which no longer requires a 2.0 percent flow uncertainty for the loss-of-coolant accident (LOCA) analysis.

The supplemental letters, References 2 through 4, contained clarifying information only, and did not change the initial no significant hazards consideration determination, or expand the scope of the initial application.

2.0 BACKGROUND

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

The uncertainty of calculating values of core thermal power determines the probability of exceeding the power levels assumed in the design-basis transient and accident analyses. In this regard, to allow for uncertainties in determining thermal power (e.g., instrument measurement uncertainties), Appendix K to 10 CFR Part 50 requires LOCA and emergency core cooling system (ECCS) analyses to assume that the reactor had operated continuously at a power level at least 102 percent of the licensed thermal power. The 2 percent power margin uncertainty value was intended to address uncertainties related to heat sources in addition to

instrument measurement uncertainties. Later, the U.S. Nuclear Regulatory Commission (NRC or the Commission) concluded that, at the time of the original ECCS rulemaking, the 2 percent power margin requirement appeared to be based solely on considerations associated with power measurement uncertainty. Appendix K to 10 CFR Part 50 did not require demonstration of the power measurement uncertainty and mandated a 2 percent margin, notwithstanding that the instruments may be more accurate than originally assumed in the ECCS rulemaking.

The Commission published a final rule in the June 1, 2000, *Federal Register* (Volume 65, Number 106, Rules and Regulations, pages 34913-34921), allowing licensees to justify a smaller margin for power measurement uncertainty. Another objective of the final rule was to avoid unnecessary exemption requests by eliminating the need for licensees to obtain exemptions. The final rule gives licensees the option of applying a reduced margin between the licensed power level and the assumed power level for ECCS evaluation, or maintaining the current margin of 2 percent power. The amended rule gives licensees the opportunity to use a reduced margin if they determine that there is a sufficient benefit. Licensees may apply the margin to gain benefits from operation at higher power, or the margin could be used to relax ECCS-related TSs (e.g., pump flows). Another potential benefit could be in modifying fuel management strategies (e.g., possibly by altering core power peaking factors). However, the final rule, by itself, does not allow increases in licensed power levels. Because the licensed power level for a plant is a TS limit, proposals to raise the licensed power level must be reviewed and approved under the license amendment process.

Caldon Incorporated (Caldon or Caldon, Inc.) Topical Report (TR) ER-80P (Reference 5), describes the theory, design, and operating features of the LEFM system. The staff approved the TR in its Safety Evaluation Report (SER) dated March 8, 1999 (Reference 6). Subsequently, Caldon supplemented the TR with Caldon ER-157P, (Reference 7). The staff approved Reference 7 in its SER dated December 20, 2001 (Reference 8). The licensee's submittal provides a plant-specific justification for a proposed 1.5 percent power uprate at Waterford 3 on the basis of References 5 and 7. This SER addresses the licensee's plant-specific justification for a 1.5 percent power uprate.

3.0 EVALUATION

The review of the application was done by various technical disciplines, and the Safety Evaluation (SE) is arranged by areas of review, as listed below:

Section 3.1	Transient and Accident Analyses Evaluation
Section 3.2	Radiological Consequences
Section 3.3	Pressure Vessel Fluence Evaluation
Section 3.4	Component and Structural Integrity
Section 3.5	Systems, Structures, and Components Evaluation
Section 3.6	Balance of Plant Systems
Section 3.7	Electrical Power
Section 3.8	Instrumentation and Control Systems
Section 3.9	Human Factors

3.1 Transient and Accident Analyses Evaluation

3.1.1 Nuclear Steam Supply System Operating Point Parameters

The NSSS Operating Point Parameters (OPPs) are the fundamental parameters used as input in the NSSS analyses. They provide the reactor coolant system (RCS) and secondary system conditions (temperatures, pressures, and flows) that are used as the basis for NSSS analyses and evaluations. As part of the requested 1.5 percent in licensed core power from 3,390 MWt to 3,441 MWt, Entergy revised these parameters. The power uprate NSSS OPPs were calculated for 1.7 percent power uprate conditions (3448 MWt) in order to bound the requested power uprate of 1.5 percent. The new parameters have been incorporated into the applicable NSSS systems and components evaluations. Additionally, the power level and initial conditions used in the Final Safety Analysis Report (FSAR) Chapter 15 analyses bound the power uprate OPPs (Reference 4, question 11).

The modified NSSS OPPs include an increased NSSS power level of 3,448 MWt and an RCS hot leg to cold leg ΔT increase of approximately 0.7 ^{o}F. Secondary side OPPs changes include an increase in feedwater temperature of 2.6 ^{o}F, an increase in feedwater and steam flowrates of approximately 2.1 percent, and a 9 pounds per square inch, absolute (psia) decrease in steam pressure. The NRC staff evaluated these changes to the plant conditions and found them to adequately represent the uprated plant behavior; therefore, the NSSS OPPs are acceptable.

3.1.2 Design Transients

The purpose of the existing design transients document is to specify the type of transients, frequency of occurrence, initial design conditions and associated thermal-hydraulic conditions experienced by various systems and components as a result of the transients. This information is then used in fatigue evaluations for those systems and components. The existing design transients represented conservative estimates for design purposes and allowed for additional margin.

To support the power increase for Waterford 3, the licensee performed a detailed evaluation which verified that the original design transients were conservatively developed with respect to the rate and extent of pressure/temperature changes during the design basis events. The most limiting normal plant transients (e.g., plant heatup and cooldown, main and auxiliary spray operation) are limited by administrative controls and/or process limits, and therefore, not impacted by the uprate. The licensee's initial evaluations of the more severe type transients (emergency, upset, and faulted conditions) were based on 102 percent reactor power (which bounds the power uprate conditions). The evaluation of the NSSS Control System Setpoint Transients demonstrated the original design transients were conservatively specified. Because the existing parameters remain bounding, the licensee concluded that the original specifications remain valid for the power uprate conditions. Additionally, the licensee concluded that the power uprate does not impact the types of transients evaluated or frequency of occurrence for any of the transients.

The NRC staff has reviewed the licensee’s conclusions regarding the impact of the power uprate on the design transients and has determined that the design transient analyses remain acceptable for the power uprate conditions.

3.1.3 Nuclear Steam Supply Systems

3.1.3.1 Reactor Coolant System

The licensee performed various assessments and demonstrated that the RCS design basis functions would be met at the revised operating conditions. Based on these assessments, the following conclusions were reached:

The major components of the main steam system support the increased heat transfer requirements. The RCS control and protection functions are not significantly affected. The RCS mass flow does not change as a result of the uprate. The RCS design temperature of 650 ^{o}F and design pressure of 2,500 psia continue to remain applicable. The pressurizer design temperature and pressure of 700 ^{o}F and 2,500 psia continue to remain applicable. And, finally, the pressurizer relief requirements increased slightly due to an increase in RCS stored energy and decay heat. However, the change is well within the relieving capacity of the pressurizer safety valves for the design transient condition.

Based on these conclusions, the RCS remains acceptable for the uprated power conditions.

3.1.3.2 Safety Injection System

The revised design conditions have no direct effect on the overall performance capability of the safety injection system (SIS). These systems will continue to deliver flow at the design basis RCS and containment pressures since there are no changes in the RCS operating pressure. The slight increase in RCS stored energy and decay heat are well within the capabilities of the SIS to respond to design basis events.

3.1.3.3 Shutdown Cooling System

The Waterford 3 shutdown cooling system (SDCS) was evaluated for an 8 percent power uprate (a possibility in the future), which bounds the requested power uprate of 1.5 percent. The evaluation consisted of normal two train and single train cooldown to cold shutdown conditions and refueling conditions, and a single train Branch Technical Position (BTP) RSB 5-1 cooldown to 200 °F. The RSB 5-1 analysis performed for the 8 percent power uprate demonstrates that the SDCS is capable of performing the required functions under the proposed 1.5 percent power uprate conditions (Reference 4, question 14). The analysis demonstrated that for the power uprate conditions, the time needed for the SDCS to reduce RCS temperature to 200 °F remains unchanged due the capability of the systems and components involved (such as emergency feedwater (EFW) flow rate, atmospheric dump valve (ADV) capacity, and SDCS capacity). Additionally, the power uprate has no effect on the SDCS flow vs. time limits in TS Surveillance Requirements 4.9.8.1 and 4.9.8.2. The staff has reviewed the results of these analyses and finds that the SDCS will continue to meet its design requirements, and is acceptable under the power uprate conditions.

3.1.3.4 NSSS Control Systems

The licensee evaluated the following transients to ensure that the plant would respond without generating a reactor trip or an engineered safety feature (ESF) actuation system (ESFAS) actuation:

- 10 percent step load decrease from 100 percent power,

- 100 percent power loss of main feedwater pump,
- 100 percent power turbine trip, and
- 5 percent per minute ramp load decrease from 100 percent power.

The licensee analyzed these transients assuming the projected operating points for a 1.7 percent power increase, which bounds the operating points for the requested 1.5 percent power uprate. Both beginning-of-cycle and end-of-cycle fuel reactivity conditions were considered. The transient analysis considered NSSS control system setpoints for the Steam Bypass Control System, the Reactor Power Cutback System, the Feedwater Control System, and the Reactor Regulating System. The results of the analysis demonstrate that the current Waterford 3 NSSS control system configuration will respond to these transients from the proposed power uprate conditions without generating a reactor trip or ESFAS actuation.

3.1.3.5 Low-Temperature Overpressure Protection Relief Valves

The Low-Temperature Overpressure Protection (LTOP) for Waterford 3 is designed to protect the RCS from overpressure events when the RCS cold legs are at temperatures less than or equal to 272 °F. The LTOP protection for Waterford 3 is provided by the two SDCS suction line relief valves in conjunction with specific operating controls.

The increase in core power due to the uprate will have a corresponding effect on decay heat, which is an input to the energy addition transient analyzed for Waterford 3. The licensee has stated that the increased heat will cause the relief valves to reach their opening pressure slightly earlier in the transient; however, due to their excess capacity, there will be no increase in peak pressure for this transient. The licensee provided additional quantitative justification in their response to the staff request for additional information (RAI) on this issue (Reference 4, question 12). The staff has reviewed the basis for this conclusion and finds that the LTOP system is acceptable under the power uprate conditions.

3.1.3.6 Core Protection Calculators

The Core Protection Calculator System (CPCS) initiates the Low Departure from Nucleate Boiling (DNB) Ratio (DNBR) and High Local Power Density (LPD) trips as well as auxiliary trips on temperature, pressure, axial shape index and radial peaking factor ranges, a variable overpower trip (VOPT) and an asymmetric steam generator (SG) transient protection trip. The increase in rated thermal power (RTP) to 3,441 MWt will require changes to the CPCS constants that set the core average heat flux and core average linear heat rate for the various algorithms. The licensee has stated that the methodology used to calculate the values for these constants is consistent with past practice and NRC-approved methods (Reference 4, question 13). The staff finds this to be acceptable because NRC approved methods are being used.

The VOPT constants are defined relative to the RTP. These constants will be changed such that the credited reactor trip will remain at the same absolute power as prior to the power uprate. Therefore, transients that rely on the VOPT will not be impacted by the uprate. These trips are addressed in Section 3.1.4.4 of this SE.

For situations where the Caldon LEFM ✓+™ system is out of service and the existing venturi-based feedwater and steam flow system must be relied upon, the licensee has stated that an appropriate penalty is applied to the calculated power such that the core protection calculation margins to trip will remain conservative (Reference 4, questions 13 and 30).

The NRC staff has reviewed the licensee's proposed changes to the CPCS constants and find them to be acceptable for the power uprate conditions because NRC-approved methods are being used and because the changes are consistent with the revised FSAR accident and transient analyses.

3.1.3.7 Core Operating Limit Supervisory System and Power Measurement Uncertainty

The Core Operating Limit Supervisory System (COLSS) consists of process instrumentation and algorithms implemented on the Plant Monitoring Computer (PMC). Although not a system required for safety, COLSS continually monitors the TS limiting conditions for operations (LCOs) on the following: linear heat rate, margin to DNBR, total core power, azimuthal tilt, and axial shape index. COLSS database constants are updated during each refueling outage to account for the changed core design. The COLSS constants that are based on the RTP will be modified to reflect the increased core power as part of the reload fuel design process.

For situations where the Caldon LEFM ✓+™ system is out of service and the existing venturi-based feedwater and steam flow system must be relied upon, the licensee has stated that an appropriate penalty is applied to the calculated power, which obviates changes to the COLSS constants by ensuring operation at an actual power level below the licensed power limit (Reference 4, question 13).

The NRC staff has reviewed the licensee's proposed changes to the COLSS constants and finds them to be acceptable for the power uprate conditions because NRC-approved methods are being used and because the changes are consistent with the revised FSAR accident and transient analyses.

3.1.3.8 Thermal Hydraulic Systems Evaluations

- Core Bypass Flow Calculation

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is, therefore, not considered effective in the core heat transfer process. The design core bypass flow limit is 2.6 percent of the total reactor vessel flow. This value was used in the thermal margin calculations. Minimizing the bypass flow maximizes the core flow, which produces higher core pressure drops and consequently, higher uplift and differential pressure loads. The licensee used a lower bound core bypass flow of 1.5 percent of reactor vessel flow in the hydraulic loads calculation. The best estimate core bypass flow is 2.28 percent of the reactor vessel flow. This is conservative for the power uprate conditions.

- Control Element Assembly Drop Time Analyses

Control Element Assembly (CEA) drop times are explicitly confirmed to meet the times assumed in the accident analyses through TS requirements. An evaluation was performed to

demonstrate continued compliance with the TS requirements. The power uprate will not significantly increase the discharge burnup for the fuel assemblies and the range of burnups and exposures of those fuel assemblies containing CEAs will remain within the current experience base (Reference 4, question 29). Furthermore, the licensee stated that since fluence-induced changes in grid cage structures will not be affected by the uprate and the fluid density has not increased significantly, the CEA drop times are not adversely affected by the uprate. Additionally, the TS requirements will continue to ensure that the times assumed in the accident analyses are satisfied.

3.1.3.9 Steam Generator Thermal-Hydraulic Performance

The licensee considered the impacts of the power uprate on the SG with respect to circulation ratio/bundle liquid flow, damping factor, pressure drop, and moisture carryover. The licensee concluded that the power uprate has negligible, if any, adverse impacts on the SG thermal hydraulic performance. Additionally, the FSAR Chapter 15 accident and transient analyses are already performed at a power level of 102 percent and implement the corresponding SG thermal hydraulic characteristics (Reference 4, question 15). The staff has reviewed the basis for the licensee's conclusion and finds it to be acceptable.

3.1.3.10 Fuel Assembly

The Waterford 3 16x16 fuel design was evaluated to determine the impact of the 1.5 percent uprate on the fuel assembly structural integrity. The evaluation demonstrated that the significant operating parameters used in the Analyses of Record bound the parameters associated with the uprate. Therefore, the fuel assembly structural integrity is not affected by the 1.5 percent uprate.

3.1.4 Nuclear Steam Supply System/Balance of Plant Fluid Systems Interface

3.1.4.1 Main Steam Safety Valves

The main steam safety valves (MSSVs) must have sufficient capacity such that the pressure does not exceed 110 percent of the SG shell-side design pressure for any pressure transients anticipated to arise. Based on this requirement, Waterford 3 has twelve MSSVs (six on each main steam line) with a minimum total capacity of 15.83×10^6 pounds per hour (lb/hr), at the highest safety valve setpoint plus accumulation pressure. This provides about 103.8 percent of the maximum calculated steam flow of 15.253×10^6 lb/hr for the power uprate design conditions.

On the basis of its review of Section 3.7.1.1, Reference 1, the staff concludes that, based on the range of the NSSS performance parameters for the uprating, the capacity of the installed MSSVs meets the sizing criteria and is, therefore, acceptable.

3.1.4.2 Power Operated Atmospheric Dump Valves

The ADVs provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when the main steam isolation valves (MSIVs) are closed or the condenser is not available. Waterford 3 has two ADVs (one on each main steam line) which are installed

upstream of the MSIVs. Each ADV is sized to have a capacity equal to approximately 5 percent of the steam flow used for plant design, at a steam pressure of 900 psia. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the EFW system. The ADVs have a total design capacity of 1.6 x 10^6 lb/hr at 885 pounds per square inch, gauge (psig). For the power uprate conditions, the ADVs capacity is approximately 10.5 percent of the required maximum steam flow. Additionally, the ADVs were designed to provide a means of decay heat removal and plant cooldown from a steady state power of 102 percent of RTP. Since the design capacity of the installed ADVs meet the sizing criterion and the ADVs were designed for 102 percent of RTP (which bounds the power uprate), the valves are adequately sized for the 1.5 percent uprated conditions. However, since this system does not perform any safety-related function, a detailed review of the impact of plant power uprate operations on the design and performance of this system was not done.

3.1.4.3 Main Steam Isolation Valves

The MSIVs function to prevent the uncontrolled blowdown of more than one SG and to minimize the RCS cooldown to within acceptable limits following a main steam line break (MSLB). The Waterford 3 MSIVs closure analysis assumed a maximum differential pressure corresponding to the second SG safety valve setpoint plus a 3 percent tolerance (Reference 4, question 16). This differential pressure is conservative and bounds the 1.5 percent power uprate conditions. Therefore, the proposed power uprate has no impact on the capability of the MSIVs to perform their design function. The staff concludes that the MSIVs meet the sizing criteria and, are therefore, acceptable.

3.1.4.4 Steam Bypass and Control System

The steam bypass and control system, located downstream of the MSIVs, creates an artificial steam load by dumping steam ahead of the turbine valves to the main condenser. The steam dump is sized to discharge 65 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction from any power level without tripping the reactor or opening the MSSVs. There are six parallel air-operated angle valves, with a total capacity of 70 percent of the uprated steam flow of 15.253 x 10^6 lb/hr, at a full load SG pressure of 831.5 psia, based on the sizing criteria of 65 percent of steam flow. Based on this, the staff concludes that the steam bypass and control system is adequate and, therefore, acceptable.

3.1.4.5 Emergency Feedwater System, Condensate Storage Pool, and Wet Cooling Tower Basin Requirements

The EFW System provides cooling water to one or both SGs for the purpose of removal of decay heat from the RCS in response to any event causing a low SG level coincident with the absence of a low pressure trip. The EFW pumps for Waterford 3 take suction from the Condensate Storage Pool (CSP) and can be aligned to the wet cooling tower (WCT) Basins. The CSP, with the minimum TS-required volume plus makeup from one WCT Basin, ensures that sufficient water is available to cool the RCS to shutdown cooling entry conditions following any design basis accident.

The licensee states that it performed BTP RSB 5-1 analysis for the 8 percent power uprate, and the original accident analysis was performed at 102 percent of RTP. The 1.5 percent power uprate will remain bounded by these analyses.

Based on review of the licensee's evaluation and experience gained from the review of power uprate applications for similar pressurized water reactor (PWR) plants, the staff concludes that the EFW, CSP, and WCT Basin are acceptable for the plant power uprate operations.

3.1.5 Nuclear Steam Supply System Accident Evaluation

3.1.5.1 Plant Protection System Setpoints

The Waterford 3 Plant Protection System (PPS) is comprised of an ESFAS and a Reactor Protection System (RPS). The ESFAS consists of the components necessary to monitor selected NSSS and containment conditions in order to generate signals to actuate the ESF and ESF support systems. The RPS is that portion of the PPS that generates signals that actuate a reactor trip.

The licensee concluded that no changes to the TS PPS setpoints are required to support the power uprate. However, two of the PPS setpoints are impacted by the power uprate because they are based on a percentage of RTP. Their values will increase slightly as a result of the power uprate. These two setpoints are discussed here.

- Linear Power Level-High

The linear power level-high trip is generated to provide reactor core protection against reactivity excursions. This trip function is not explicitly credited in any of the FSAR Chapter 15 analyses. As such, the analysis limit in the TS is not changed.

- Logarithmic Power Level-High

The logarithmic power level-high trip is generated to protect the integrity of the fuel cladding and the RCS pressure boundary in the event of an unplanned criticality from a shutdown condition (e.g., CEA withdrawal from subcritical). In Reference 4, question 18, the licensee discussed the impact of the power uprate on this setpoint. The Cycle 10 (to present) analyses have used a trip setpoint of 4.4 percent of RTP. The actual high logarithmic power trip setpoint is less than or equal to 0.257 percent RTP (TS 2.2.1). The total setpoint plus uncertainty associated with these instruments is less than 1 percent RTP. In addition, accounting for potential decalibration effects would increase the analysis setpoint to a value of 2.08 percent RTP. The trip setpoint modeled in the Waterford 3 analyses are conservative and bounding with respect to this value. Thus the accident consequences remain bounding and applicable for the 1.5 percent power uprate. No TSs for this setpoint are necessary.

The staff has reviewed the impact of the Waterford 3 power uprate on the PPS setpoints and agrees that the current setpoints remain valid and no TSs are necessary.

3.1.5.2 Emergency Core Cooling System Performance

The Waterford 3 ECCS performance analysis consists of three analyses, the Large Break LOCA (LBLOCA), the Small Break LOCA (SBLOCA), and post-LOCA Long Term Cooling (LTC) analyses. Consistent with the requirement of Paragraph I.A of Appendix K to 10 CFR Part 50, these analyses are performed assuming a core thermal power level equal to or greater than 102 percent of the current licensed power level of 3,390 MWt.

As allowed by the recent revision to Paragraph I.A of Appendix K, this amendment request proposes to increase the licensed core power level by 1.5 percent to 3,441 MWt and to decrease the power measurement uncertainty to no greater than 0.5 percent. The proposed 1.5 percent power uprate takes advantage of the reduced power measurement uncertainty without changing the analyses initial power assumptions.

In Reference 4, question 32, the licensee stated that the LBLOCA and SBLOCA analysis methodologies presently approved for Waterford 3 continue to apply by providing a statement that Waterford 3 and its vendor(s) have ongoing processes which assure that LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as-operated plant values for those parameters.

Waterford 3 has demonstrated that the LBLOCA and SBLOCA analysis methodologies presently approved continue to apply. Therefore, we find that the LOCA analyses presently approved for Waterford 3 continue to apply and are suitable for inclusion in plant licensing documentation, including the TS and core operating limits report.

Regarding the Waterford 3 LTC analysis, in Reference 4, question 31, the licensee stated that its LTC analysis of record was performed at 102 percent of RTP, and that this bounds the requested power uprate condition since the analysis was not revised for the 1.5 percent power uprate. In a number of power uprate reviews, the staff has found that a correctly performed LTC analysis based on 102 percent power is applicable to a power uprate when the 2 percent margin is reapportioned between the proposed new power and a justified change in the power determination uncertainty, since this is consistent with the recent change in Section I.A of 10 CFR Part 50, Appendix K.

The existing analyses of record are acceptable for the requested thermal power level because; (1) the licensee uses a previously approved methodology, (2) the total of the requested thermal power level plus instrument uncertainty remains unchanged, and (3) there are no other changes in the analyses of record.

The staff is conducting a review of boric acid accumulation modeling and determination of hot-leg switchover time. The issues include modeling of the decay heat generation rate, the appropriate margin for calculation of boric acid concentration, and determination of a conservative volume in which boric acid accumulates. These issues apply to a number of licensees, they are not safety-significant, and the staff is confident that a satisfactory resolution will not have a significant impact. Consequently, the issues are being addressed generically and, pending resolution, licensee's analyses of record continue to be acceptable.

3.1.5.3 Non Loss-of-Coolant Accident/Transient Analyses

The Waterford 3 non-LOCA transient analyses assume an initial power level of 102 percent of RTP (includes a 2 percent power uncertainty) for those events in which higher power produces more adverse results. The majority of transients are either analyzed at 102 percent of RTP or are bounded by events which are analyzed at 102 percent of RTP. As allowed by the recent revision to Paragraph I.A of Appendix K, the licensee proposes to increase the licensed core power level by 1.5 percent to 3,441 MWt, and reduce the power uncertainty to less than 0.5 percent. Consequently, the current non-LOCA transient analyses remain bounding for the uprated power level.

In order to demonstrate that the non-LOCA transient analyses are indeed bounding for the power uprate conditions, the licensee provided a matrix of the FSAR Chapter 15 transients including discussions of Standard Review Plan (SRP) (Reference 9) acceptance criteria and impacts of the power uprate on the transient results. The licensee provided verification (Reference 4, question 22) that:

a. The analyses of record were either previously approved by the NRC or were conducted using methods and/or codes that were previously approved by the NRC.
b. The analyses contained in FSAR Revision 11 are the current analyses of record. For the CEA Withdrawal from Subcritical, CEA Withdrawal from Hot Zero Power, CEA Ejection, and MSLB events, the current analysis of record was submitted as part of References 10 and 11.
c. The bounding event determinations remain bounding.

Of the FSAR Chapter 15 non-LOCA transients, only the following events are analyzed at a RTP less than 102 percent of 3,390 MWt:

- 15.1.3.2 - Steam System Piping Failures MODES 3 and 4 with all Full Length CEAs on the Bottom - This analysis is initiated from subcritical conditions and, as such, the power uprate has no impact on this transient.
- 15.4.1.1 - Uncontrolled CEA Withdrawal from Subcritical - This analysis is initiated from subcritical conditions and was re-analyzed as part of the Reference 10 supplemental amendment request. The power uprate has no impact on any of the acceptance criteria for this event. During the power uprate review, it was identified that the TS Peak Linear Heat Rate (PLHR) Safety Limit was being violated for this event, prompting the licensee to change this safety limit to a fuel centerline melt temperature safety limit. This issue was resolved as part of the Reference 12 amendment request.
- 15.4.1.2 - Uncontrolled CEA Withdrawal from Low Power - This analysis is initiated from low power conditions and was re-analyzed as part of the Reference 10 supplemental amendment request. The power uprate has no impact on any of the acceptance criteria for this event. During the power uprate review it was identified that the TS PLHR Safety Limit was being violated for this event, prompting the licensee to change this safety limit to a fuel centerline melt temperature safety limit. This issue was resolved as part of the Reference 12 amendment request.
- 15.4.1.3 - Uncontrolled CEA Withdrawal at Power - Based on parametric analyses covering a range of initial core power levels, the initial core power level assumed in this analysis is 76 percent of RTP. This actuates a DNBR trip just prior to the high pressurizer pressure trip and the high power level trips, which maximizes the rate of approach to the fuel design limits and produces the most severe consequences. The

DNBR trip setpoint is not affected by the power uprate and, therefore, the event consequences remain bounding (Reference 4, question 25).

- 15.4.1.4 - CEA Misoperation - This event is initiated from a Power Operating Limit of 3,410 MWt. The sets of initial conditions (power, pressure, temperature, coolant flow rate, radial peaking factor, and axial power distribution) that produce the most adverse consequences following CEA misoperation events were determined by performing parametric analyses of each incident over the range of reactor operating conditions, including reactor power less than or equal to 102 percent of 3,410 MWt, which bounds the uprate power of 3,441 MWt.
- 15.4.1.7 - Uncontrolled CEA Withdrawal from a Subcritical Condition with all Full Length CEAs on the Bottom - In Reference 4, question 27, the licensee reported that this event is initiated from subcritical conditions (MODES 3, 4, and 5) and is bounded by the Uncontrolled CEA Withdrawal from Subcritical event discussed above. The power uprate has no impact on this event.
- 15.4.3.1 - Inadvertent Loading of a Fuel Assembly into an Improper Position - For those mis-loading events which would not be detected during startup testing, the licensee stated that the events would not be adversely impacted by the power uprate. In Reference 4, question 28, the licensee provided adequate justification, showing that the heat flux used in the FSAR calculation is sufficient to bound the uprate power and that the power uprate will not significantly affect either the power distributions or the relative fuel assembly reactivities.

In reviewing Reference 1, it was identified that certain non-LOCA transients, as described in FSAR Chapter 15, show Minimum DNBR (MDNBR) results of 1.19, which is the CE-1 DNB correlation limit. The current Waterford 3 MDNBR limit is based on the Modified Statistical Combination of Uncertainties (MSCU) methodology and is 1.26. A telephone conference with the licensee confirmed that all transients for which the MSCU method applies are evaluated using that method and the current 1.26 MDNBR acceptance criteria. Because the consequences for the events analyzed using the Deterministic method with a MDNBR of 1.19 remained bounding, the licensee did not update the FSAR. In accordance with the requirements of 10 CFR 50.71, the licensee should update the FSAR to reflect this in the next FSAR revision.

The staff has reviewed the information submitted by the licensee (References 1 and 4), including a review of the initial power level assumptions, bounding event discussions, and transient results, as described in the FSAR and the Reference 10 and 11 submittals. The staff has concluded that, because the total core power level assumed in the non-LOCA transient analyses and all other analysis inputs bound the power uprate nominal operating points (Reference 4, question 11) and because all transients meet their acceptance criteria, the non-LOCA transient analyses are acceptable for the uprated power conditions.

3.1.5.4 Trip Setpoints

The power uprate will have an impact on the trip setpoints, which are based on a percentage of the RTP. The reactor trips that are based on percentage of RTP are:

1. High Logarithmic Power Trip
2. High Linear Power Trip

3. CPCS Variable Overpower Trips

    a. CPCS VOPT Setpoint Variable Minimum Value (SPVMIN)
    b. CPCS VOPT Setpoint Variable Maximum Value (SPVMAX)
    c. CPCS VOPT “Rate of Change” (SUPMAX, SDNMAX)
    d. CPCS VOPT “Offset” (DELSPV)

Trips 1 and 2 above are part of the RPS and were addressed in Section 3.1.5.1 of this SE. The trips listed in item 3 above are auxiliary trip functions of the CPCS. Although the Core Protection Calculators are part of the Reactor Protection Instrumentation and are listed in TS Table 3.3-1, the CPCS VOPT setpoints discussed here are not specifically TS setpoints. However, because they are based on a percentage of RTP and are being changed, they are discussed here:

a. Core Protection Calculator System Variable Overpower Trips/Setpoint Variable Minimum Value

SPVMIN, the floor of the VOPT, is used as mitigating action against transients starting from a low power state. Currently this is set to 30 percent of RTP: 3,990 MWt. To maintain the credited reactor trip at the same absolute power level, Waterford 3 will reduce SPVMIN by the ratio of the new and old RTP definitions. Thus, for operation at a RTP of 3,441 MWt, SPVMIN will have a setpoint of 29.6 percent of 3,441 MWt. Based on this change, the FSAR Chapter 15 analyses which credit this trip will not be impacted by the power uprate.

b. Core Protection Calculator System Variable Overpower Trips/Setpoint Variable Maximum Value

SPVMAX, the maximum value of the VOPT, is a high power trip setpoint. Currently this is set to 110 percent of RTP: 3,990 MWt. To maintain the same relationship between the initial conditions and the trip setpoint in terms of absolute power changes, SPVMAX will be reduced by the ratio of the new and old RTP definitions. Thus, for operation at a RTP of 3,441 MWt, SPVMAX will have a setpoint of 108.3 percent of 3,441 MWt. Based on this change, the FSAR Chapter 15 analyses which credit this trip will not be impacted by the power uprate.

c. Core Protection Calculator System Variable Overpower Trips - “Rate of Change” (SUPMAX, SDNMAX)

At steady state power conditions, the trip setpoint is set 8 percent above the existing power. There is a maximum rate at which the trip setpoint can increase as core power starts to increase during transients. This maximum rate of increase, SUPMAX, is currently set to 2 percent/minute. To maintain the same relationship between the transient conditions and the trip setpoint in terms of absolute power changes, SUPMAX will be reduced by the ratio of the new and old RTP definitions. Thus, for operation at a RTP of 3,441 MWt, SUPMAX will have a setpoint of 1.97 percent of 3,441 MWt/minute. The maximum rate of decrease of the setpoint, SDNMAX, will not be changed since the transient analysis is not sensitive to its value. Based on this change, the FSAR Chapter 15 analyses which credit this trip will not be impacted by the power uprate.

d. Core Protection Calculator System Variable Overpower Trips - “Offset” (DELSPV)

The VOPT setpoint is set by an offset above the steady state power level. This offset, DELSPV is currently set to 8 percent of 3,390 MWt above the initial power at the start of the transient. This trip moves at a prescribed rate as the transient progresses. The trip is limited to the range of SPVMIN and SPVMAX. To maintain the same relationship between the initial conditions and the trip setpoint in terms of absolute power changes, DELSPV will be reduced by the ratio of the new and old RTP definitions. Thus, for operation at a RTP of 3,441 MWt, DELSPV will have a setpoint of 7.8 percent of 3,441 MWt/minute. Based on this change, the FSAR Chapter 15 analyses which credit this trip will not be impacted by the power uprate.

The staff has reviewed these CPCS VOPT setpoint changes and find them to be acceptable because FSAR Chapter 15 analyses which credit these VOPT setpoints will not be impacted by the power uprate.

3.1.5.5 Steam Generator Tube Plugging

The power uprate will have no impact on the tube plugging assumptions used for the Waterford 3 FSAR Chapter 15 analyses. The tube plugging assumptions used in the current accident analyses are based on a range of tubes plugged, from 0 up to 500 tubes plugged per SG. Events that are limiting at either extreme of the plugged tube spectrum have already been analyzed at the uprated power.

3.1.6 Nuclear Fuel

3.1.6.1 Fuel Core Design

The effects of the power uprate on the fuel core design were evaluated using the current design for the upcoming fuel cycle (Cycle 12) and the currently planned cycles numbered 13 through 16. The licensee stated first that the methods and core models used in the uprate analyses are consistent with those presented in the Waterford 3 FSAR. Secondly, they claim that the core analyses show that the uprate does not result in changes to the current nuclear design basis documented in the FSAR. And finally, the licensee asserts that the impact on the peaking factors, rod worth, reactivity coefficients, shutdown margin, and kinetics parameters are either well within normal cycle-to-cycle variation or are controlled by the core design and will be addressed on a cycle-specific basis consistent with the reload methodology. The licensee provided a listing of the Waterford 3 nuclear fuel core design, and thermal-hydraulic and fuel rod design codes and methodologies in Reference 4, question 21, and also confirmed that processes are in place such that all analyses employed in support of the power uprate satisfy applicable SER code limitations and constraints.

The staff finds the licensee’s analyses and controls to be acceptable for the power uprate with the current fuel core design because NRC-approved codes and methodologies are used, and processes are in place to ensure that any code limitations and constraints imposed by NRC SERs are satisfied.

3.1.6.2 Core Thermal-Hydraulic Design

The licensee evaluated the core thermal-hydraulic design and methodology at the uprated core power level of 3,441 MWt. The Waterford 3 thermal hydraulic design is based on the TORC computer code, the CE-1 Critical Heat Flux correlation, the simplified TORC modeling methods and the CETOP computer code. In addition, the DNBR analysis uses the methodology for determining the limiting fuel assembly(ies). To validate the DNBR design limit of 1.26, the licensee applied the MSCU. The licensee stated that the core thermal-hydraulic design and methodology remain applicable at the uprated core power level of 3,441 MWt.

The staff finds the licensee's core thermal-hydraulic design and methodology to be acceptable for the power uprate because NRC-approved codes and methodologies are used, and processes are in place to ensure that any code limitations and constraints imposed by NRC SERs are satisfied.

3.1.6.3 Fuel Rod Design

The licensee evaluated the thermal performance of erbia and UO_2 fuel rods for a 1.5 percent power uprated Waterford 3 core. The evaluation included a power history that enveloped the power and burnup levels expected for the peak fuel rod at each burnup interval, from beginning-of-cycle to end-of-cycle burnups. The licensee found that the maximum predicted fuel rod internal pressure remains below the critical pressure for the No-Clad-Lift-Off criterion. In Reference 4, question 20, the licensee also demonstrated acceptable corrosion performance for the planned Waterford 3 1.5 percent power uprate core.

Since the fuel rod design continues to meet all of the acceptance criteria requirements, we find the design to be acceptable for the power uprate.

3.1.7 Anticipated Transients Without Scram

To comply with 10 CFR 50.62, the Anticipated Transients Without Scram (ATWS) mitigation system actuation circuitry has previously been incorporated into the design of Waterford 3. The Waterford 3 mitigating systems include diverse reactor trip, diverse turbine trip, and diverse EFW actuation systems. The licensee has evaluated the ATWS mitigation system with respect to the proposed 1.5 percent power uprate and determined that no changes are needed. The staff agrees with the licensee's assessment and concludes that Waterford 3 will continue to comply with 10 CFR 50.62 for the uprated power conditions.

3.1.8 Summary

The staff reviewed the power uprate amendment in the areas of Transient and Accident Analyses. The staff finds Entergy's request to permit operation at 3,441 MWt for Waterford 3 to be acceptable. This acceptance is based on the results of the review of the submitted information and the acceptance criteria of the staff SRP (Reference 9). The staff has confirmed that the licensee used analytical methods that have been previously reviewed and approved by the NRC and that all acceptance criteria continue to be met.

3.2 Radiological Consequences

3.2.1 Evaluation

The staff reviewed the impact of the proposed changes on design basis accident radiological analyses, as documented in Chapter 15 of the Waterford 3 FSAR. The staff used the information currently contained in the Waterford 3 FSAR, in addition to that submitted by the licensee in support of the requested license amendment. The licensee stated in Reference 1 that the source terms used in the fuel handling accident (FHA), the LOCA and the Maximum Hypothetical Accident (MHA) are based on 102 percent of the current rated core power level of 3,390 MWt. For non-LOCA design basis accidents that do not assume fuel failure, the source terms are based on the TS allowable RCS activity limits with the assumed steam activity release rates based on at least 1.02 times the licensed core thermal power. The 2 percent margin included in these analyses bound the 1.5 percent power uprate conditions.

For non-LOCA design basis accidents that assume fuel failure occurs, the current FSAR analysis source terms are based on a maximum radial peaking factor and a core power of 3,390 MWt (100 percent of the current rated core power). The licensee stated in the submittal that since current design constraints limit the hot rod radial peaking factor to lower than the maximum assumed in the design basis accident analyses, the current non-LOCA source term will be applicable up to the 1.5 percent uprate conditions. The staff asked for further information on this point, which was received by Reference 2. The staff found the additional information supported the licensee's assertion that the fuel-failure non-LOCA source terms remain bounding for the power uprate. After the power uprate has been implemented, any recalculation done by the licensee of the radiological effects of design basis non-LOCA events that assume fuel failure, should use a source term based on the current rated core thermal power at that time. The staff finds that the assumptions discussed above and the current FSAR design basis accident radiological analyses bound the expected conditions after the proposed power uprate.

3.2.2 Summary

Considering the information provided by the licensee, as well as information documented in the Waterford 3 FSAR, and based on the above discussion, the staff finds that the current FSAR Chapter 15 radiological analyses remain bounding for the proposed 1.5 percent power uprate. The staff finds that the proposed changes are acceptable with respect to the radiological consequences of design basis accidents.

3.3 Pressure Vessel Fluence Evaluation

3.3.1 Background

By letter dated December 14, 1993, (Reference 13), Entergy submitted the required information to establish pressure-temperature (PT) limits for 20 Effective Full Power Years (EFPYs). However, Entergy was not able to provide a timely response to the staff's requests for information regarding fluence uncertainty for the next scheduled outage and, by letter dated March 3, 1995, (Reference 14), requested to shorten the extension to 15 EFPYs. The staff estimated that the difference was more than adequate to cover the potential uncertainties and approved the extension to 15 EFPYs. By letter dated July 15, 1999, (Reference 15), Entergy proposed to extend the PT curves from 15 to 20 EFPYs. However, by letter dated January 6, 2000, (Reference 16), the licensee revised the July 15, 1999, request for an extension to 16 EFPYs. The staff estimated that there was sufficient margin to cover potential fluence

uncertainties for a one year extension (from 15 to 16 EFPYs) and the extension was approved. Finally by Reference 1, Entergy proposed a 1.5 percent power uprate and requested that the 16 EFPY PT limit be acceptable for the power uprated core. The purpose of this review is to establish that there is sufficient margin to operate through Cycle 12, which will complete the available 16 EFPYs.

At the end of Cycle 11, the licensee will remove the second surveillance capsule. The results of the capsule analysis will become available in about one calendar year. At the end of Cycle 11, the plant will be at 14 EFPYs and, at the end of Cycle 12, will be at 16 EFPYs.

3.3.2 Evaluation

Even though the original fluence estimate was performed for 20 EFPYs, the cross sections differed from the guidance of Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." However, because Waterford 3 does not have a thermal shield, the effect of the old cross sections set in the calculated fluence is minimized. Further, the revised cross sections primarily affected iron, and penetration through a thermal shield could yield an underestimation of the fluence by as much as 20 percent. However, the deficit is about 5 percent in the absence of a thermal shield. There are two sources of conservatism that support the licensee's request, 1) the staff estimates that a 20 percent margin is present since the calculated value was originally estimated for 20 EFPYs, while the PT limits expire at 16 EFPYs, and 2) the licensee has been using low leakage (in-out) core loadings for several cycles, while the 20 EFPY estimate was based on (out-in) loadings, which maximize neutron leakage and fluence values. The licensee did not quantify these savings, but staff experience indicates that it is, at a minimum, several percent, while the power uprate will increase the 16 EFPY fluence by about one percent. Therefore, the staff concludes that there is sufficient margin to operate to 16 EFPYs with the current PT curves and LTOP protection limits.

The licensee will revise the fluence for Cycle 13 (and subsequent cycles) after the evaluation of the second surveillance capsule. This capsule is expected to capture the effect of the low leakage loadings, as well as the power increase. The analytical results will be calculated with codes adhering to the guidance of RG 1.190, thus, removing the question of fluence uncertainty. There are no TS changes associated with the proposed extension. The licensee also concluded that the existing Waterford 3 thermal shock and upper shelf energy evaluations will remain acceptable until such time as the evaluation of the next surveillance capsule is complete. The NRC staff reviewed the licensee's analysis and reasoning, and finds it acceptable.

3.4 Component and Structural Integrity

3.4.1 Chemical and Volume Control System

The primary role of the chemical and volume control system (CVCS) is to provide for boric acid addition and removal, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, and processing of reactor coolant letdown.

During plant operation, reactor coolant letdown is taken from the cold leg on the suction side of the reactor coolant pump (RCP), through the tube side of the regenerative heat exchanger, and then through letdown control valves. The regenerative heat exchanger reduces the temperature of the reactor coolant and the control valves reduce the pressure. The letdown is cooled further in the tube side of the letdown heat exchanger and subsequently passes through the purification filter. Flow continues through the purification ion exchangers, where ionic impurities are removed, and enters the volume control tank (VCT). The charging pumps take suction from the VCT and return the coolant through the shell side of the regenerative heat exchanger to the RCS in the cold leg, downstream of the RCP. The nominal T_{COLD} for the power uprate remains unchanged at 545.0 ^{o}F. As a result, the temperature of the letdown flow is not changed. The licensee concluded that there is no impact on the thermal performance of the CVCS as a result of the proposed power uprate.

The CVCS also provides a source of borated water for post accident injection. Evaluation of required ECCS water volumes and boric acid concentrations will be performed as part of the normal reload SE process. The slight increase of N-16 activity at uprate conditions has a negligible effect on letdown line delay time requirements. Therefore, no change to the letdown and makeup requirements as a result of power uprate is expected. The licensee concluded that the cold leg temperature (T_{COLD}) and the reactor coolant mass flow rate remain unchanged. Increased power is due to a slight increase in the hot leg temperature (T_{HOT}) and associated increase in the average temperature (T_{AVG}). The increase in T_{AVG} will cause a small increase in the makeup requirements for coolant shrinkage during cooldown. However, this effect is considered negligible. The staff reviewed the licensee's analysis and concurs with this conclusion.

3.4.2 Steam Generator Structural Integrity

Waterford 3 has Combustion Engineering model 3410 SGs which have mill-annealed Alloy 600 tubes.

The licensee evaluated the structural integrity of the Waterford 3 SG tubes under the following loading conditions, as specified in RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes":

- the normal operating differential pressure,
- 1.4 times the MSLB differential pressure, and
- 3 times the normal operating differential pressure.

The normal operating differential pressure of 1402 psi (pounds per square inch) has been adjusted for the power uprated conditions to 1420 psi. In addition, the licensee increased the normal operating differential pressure by 10 percent to adjust for temperature and added an additional 50 psi for instrumentation error. The licensee concluded that the SG tubes under the power uprated conditions will maintain the safety margin recommended in RG 1.121.

To evaluate the leakage integrity of the SG tubes, the licensee evaluated the screening criteria for flaw evaluation considering the power uprated conditions. The screening criteria contain structural limits and flaw depth thresholds for the in-situ leak and pressure testing of degraded tubes. On the basis of its evaluation, the licensee stated that the screening criteria for the flaw

evaluation are not affected by the power uprate. The leakage integrity will not be affected significantly under the power uprated conditions.

In addition to SG tubes, the licensee evaluated the following SG structures in accordance with the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (the Code): tubesheet, primary head divider plate, baffle and baffle support, secondary shell, and feedwater nozzle.

As indicated in Table 3.3.1-1 in Reference 1, the operating condition (i.e., SG steam flow) has increased slightly (1 percent) for the 1.5 percent power uprate. The licensee performed an evaluation to determine the effects of key loading changes on the subcomponents. The licensee also evaluated SG tubes for the effect of LOCA load increases on tube degradation, effect of thermal hydraulic load changes on tube flow induced vibration, and the effect of increased primary to secondary Delta P on tubesheets and related structures. The licensee also performed an evaluation of the primary head divider plate, baffle and baffle support, secondary shell, and feedwater nozzle fatigue for the effects of uprate-driven parameter changes. On the basis of its structural evaluations, the licensee concluded that the SGs meet the ASME Code of record limits for stress and fatigue for the 1.5 percent power uprate condition. The staff finds the licensee's conclusion acceptable.

The licensee also evaluated the impact of the revised design conditions associated with the 1.5 percent power uprate on the structural integrity of the SG hardware changes and additions. The mechanical tube plugs were evaluated for the effects of changes to the thermal transients due to power uprate. Based on its evaluation, the licensee concluded that all primary stress limits are satisfied for the plug shell wall, and the plug shell continues to meet the Class 1 fatigue requirements for materials, design, fabrication, testing, and inspection of Article N-415.1 of the 1966 Edition of Section III of the ASME Code.

On the basis of its review, the staff finds the licensee's conclusion that the SGs at Waterford 3 will continue to maintain their structural and pressure boundary integrity and remain in compliance with the Code of record specified in the FSAR acceptable for the proposed 1.5 percent power uprate. The staff finds the rationale for the licensee's conclusion that the leakage integrity of the SG tubes will be maintained under the power uprate acceptable.

3.4.3 Steam Generator Hardware Changes and Additions Evaluation

Industry operating experience has shown that SG mechanical tube plugs manufactured by Westinghouse Electric Company (Westinghouse) and Framatome, using Alloy 600 material with certain heats, are susceptible to primary water stress corrosion. The NRC has issued the following generic communications regarding the tube plug degradation: Information Notices 89-33, 89-65, and 94-87, and Bulletin 89-01 with associated Supplements 1 and 2. Bulletin 89-01 recommends that all Westinghouse Alloy 600 mechanically rolled tube plugs with certain heats be removed.

In spring 1991, the licensee replaced the hot leg Westinghouse Alloy 600 mechanical tube plugs with Westinghouse Alloy 690 mechanical ribbed plugs. In spring 1997, the licensee replaced the last 11 cold leg Westinghouse Alloy 600 mechanically rolled tube plugs with

Framatome Alloy 690 long-threaded mechanical plugs. The licensee has also used Westinghouse Alloy 690 mechanical rolled plugs.

The licensee evaluated the Westinghouse ribbed and rolled plugs and Framatome rolled mechanical plugs for the effects of thermal transients under the power uprated conditions. All of the actual versus allowable stress ratios were found to be less than unity. This indicates that all primary stress limits are satisfied for the plug shell wall between the top land and the plug end cap. The plug shell continues to meet the Class 1 fatigue exemption requirements per Article N-415.1 of the 1966 Edition of Section III of the ASME Code, equivalent to NB-3222.4 of the 1989 Edition and Section III of the 1986 Edition of the ASME Code including ASME Code Case N-474-1. The fatigue exemption requirements are satisfied and, therefore, the usage factor will remain within the Code limit of 1.0.

In addition, the license evaluated the Westinghouse Alloy 690 mechanically rolled plugs for the effects of an increase in the primary to secondary pressure differential under the power uprated conditions. The evaluation shows that the Westinghouse Alloy 690 rolled plugs satisfy applicable Code requirements under the power uprated conditions.

The staff finds that the SG tube plugs will satisfy the ASME Code under the power uprated conditions.

3.4.4 Inspection Program and Tube Repair Criteria

The licensee identified the following active degradation mechanisms in the Waterford 3 SGs:

1) Wear indications at the square bend locations and upper bundle (batwing) supports,
2) Outside diameter stress corrosion cracking (ODSCC), primary water stress corrosion cracking (PWSCC), and volumetric indications at the top of the tubesheet,
3) Axial and volumetric indications at the egg-crate tube support intersections, and
4) PWSCC in the tight radius u-bend regions of tubes in rows 1 to 3.

Waterford 3 SGs have experienced wear at diagonal and vertical supports since the first cycle of operation. The licensee has preventively plugged 305 tubes in the stay cylinder regions of each SG. To date, the licensee is monitoring over 600 wear indications at the support structures. The majority of the wear indications are identified at the square bend supports. The balance of the wear indications are identified within straight sections of tubing that were located at or below the seventh hot and cold leg egg-crate support intersections. The licensee evaluated the wear indications at the support locations under the power uprated conditions and concluded that the wear indications will not be affected significantly under the power uprated conditions. The licensee stated that any additional tube that may be affected by the support wear would have relatively slow wear rates and would be removed before the structural limits were compromised.

The licensee stated that the temperature increase above the present T_{HOT} of 600 °F will affect the temperature-dependent ODSCC and PWSCC. However, the power uprate will increase primary water bulk temperature by only 1 °F. With this small temperature increase, the licensee concluded that the power uprated conditions will have negligible impact on ODSCC and PWSCC.

The licensee stated that the inspection intervals for the SG tubes will not be changed under the power uprated conditions; however, the higher temperatures in crack growth rate analyses as a result of the power uprate will be considered in the inspection program. The licensee currently inspects the Waterford 3 SGs every scheduled refueling outage. The Waterford 3 inspection program, subsequent to refueling outage 11 in spring 2002, will continue to be comprehensive and adhere to industry recommendations with regards to potential damage mechanisms and qualified inspection techniques.

The licensee stated that the primary-to-secondary pressure differential will be increased under the uprated power conditions; however, the current tube repair criterion and the 40 percent throughwall tube plugging limits specified in the plant TSs remain valid. The increase in the primary-to-secondary pressure differential will be accounted for and adjusted in the future SG degradation assessments, and incorporated in the screening criteria of the in-situ pressure testing.

The staff agrees with the licensee's conclusions that the inspection program, tube plugging limits, and tube repair criterion will not be affected significantly under the power uprated conditions.

3.4.5 Steam Generator Blowdown System

The function of the SG blowdown system is to control the chemical composition of the SG shell water to be maintained within the specified limits. The blowdown system also controls the buildup of solids in the SG water.

The blowdown flow rates required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. The rate of addition of dissolved solids to the secondary systems is a function of condenser leakage and the quality of secondary makeup water. The rate of generation of particulates is a function of erosion-corrosion (E/C) within the secondary systems. Since neither condenser leakage nor the quality of secondary makeup water is expected to be impacted by power uprate, it is not expected that the blowdown flow rate required to address dissolved solids will be impacted by power uprate. The overall effect of the minor increases in secondary system velocities is not expected to alter the E/C rates appreciably. The licensee concluded that the required blowdown flow rate to control secondary chemistry and particulates will not be significantly impacted by power uprate.

Since the inlet pressure to the SG blowdown system varies proportionally with operating steam pressure, the blowdown flow control valves must be designed to handle a corresponding range of inlet pressures. Based on the revised range of NSSS parameters for power uprate, the no-load steam pressure (1050 psia) remains the same and the full-load minimum steam pressure (831.5 psia) is within the present operating range. The licensee concluded that the range of operating parameters revised for power uprate will not impact blowdown flow control.

The NRC staff reviewed the licensee's analysis and finds it acceptable.

3.4.6 Flow Accelerated Corrosion

The licensee stated that it has committed to adhere to criteria, codes, and standards for high-energy piping systems described in current licensing basis documents. In addition, part of this commitment is to maintain piping within the allowable thickness values. The piping systems at Waterford 3 are modeled using CHECWORKS. This modeling program will be revised, as appropriate, to incorporate flow and thermodynamic states that are projected for uprated conditions. The licensee also stated that the results of these models will be factored into future inspection and pipe replacement plans consistent with current Flow Accelerated Corrosion (FAC) program requirements.

The staff requested details from the licensee regarding the revisions to the CHECWORKS model. The licensee responded that updated parameters include power level, steam flow rate, pressure, temperature, blowdown rate, carryover, and steam enthalpy. In addition, the licensee stated that the overall impact of updating the model parameters will be minimal with respect to the component inspection and replacement. Based on the information provided, the staff concludes that the proposed 1.5 percent power uprate will not adversely impact the current FAC program at Waterford 3, and that the changes to the FAC model will be adequately factored into the current FAC program to maintain piping within the allowable thickness values.

3.4.7 Summary

The staff reviewed and evaluated the proposed amendment in the areas of component and structural integrity. Based on the review of the licensee's rationale and evaluation, the staff evaluations described above, and the experience gained from our review of power uprate applications for similar PWR plants, the staff concludes that Waterford 3 operations at the proposed power uprate is acceptable.

3.5 Systems, Structures, and Components Evaluation

3.5.1 Evaluation

The staff reviewed the power uprate amendment for Waterford 3 as it relates to the structural and pressure boundary integrity of the NSSS and balance-of-plant (BOP) systems. Affected components in these systems include piping, in-line equipment and pipe supports, the reactor pressure vessel, core support structures, reactor vessel internals (RVI), SGs, CEA mechanisms (CEAM), RCPs, and pressurizer. The staff's SE concerning the effects of the power uprate on the pertinent components is provided below.

3.5.1.1 Reactor Vessel

The proposed power uprate for Waterford 3 will increase the operating core power level by approximately 1.5 percent over the currently licensed level of 3,390 MWt. The licensee reported that the power increase will result in the revised design parameters given in Table 3.3.1-1, Attachment 2 of Reference 1, for the proposed power uprate conditions. As noted in Table 3.3.1-1, the reactor pressure remains unchanged. The licensee indicated that the design basis analyses of record for the existing LOCA loads remain bounding for the proposed power uprate condition. In response to the staff’s RAI, the licensee, in Reference 4,

confirmed that the current design basis LOCA produced by the as-built tributary lines are bounded by the design basis LOCA resulting from the mechanistic failure of main coolant loop piping.

In Reference 4, the licensee provided revised Table 3.3.1-1 that includes design parameters for the original design, operating, and power uprate conditions. Based on a comparison of these parameters, it is noted that the nominal vessel inlet and outlet temperatures associated with the 1.5 percent power uprate are essentially the same as the nominal temperature for the current cycle. A slight change in the inlet and outlet temperatures is bounded by the inlet and outlet temperatures that were used in the original analysis of the reactor vessel nozzles. The licensee concluded that the effects of the plant loading and unloading transients on the inlet and outlet nozzles remain bounded by the stress analyses of record. Also, the licensee, in Section 3.4 of Reference 1, concluded that the current design basis transients will remain unchanged for the uprated condition. Therefore, the existing reactor vessel structural analyses remain bounding for the 1.5 percent power uprate condition. The licensee concluded that the stress intensities and cumulative usage factors (CUFs) of the reactor vessel components will continue to satisfy the limits in the code of record, ASME Code Section III, 1971 Edition, including the summer 1971 addenda. The staff agrees.

3.5.1.2 Reactor Core Support Structures and Vessel Internals

The licensee evaluated the reactor internal components, considering the revised design conditions provided in Table 3.3.1-1 of Reference 1. The licensee stated that, with little or no increase in flow and change in the RCS nominal operating temperatures, the change in the boundary conditions experienced by the reactor internal components is insignificant. Also, the 1.5 percent power uprate does not change the current design basis seismic and LOCA hydraulic and dynamic loads. In Reference 4, responding to one of the items in the staff's RAI, the licensee stated that the flow-induced vibration effects considered in the analysis of record are more severe than the flow-induced vibration effects associated with the 1.5 percent uprate condition. The licensee's assessment of the reactor internal structures determined that the thermal gradients and hydraulic loads are bounded either by its previous analyses performed for a proposed 8 percent uprate at Waterford 3, or by analyses performed for another reactor with the same reactor internal component configuration and characteristics.

In Reference 4, the licensee provided additional information requested by the staff with regard to the evaluation of the RVI structures. The licensee stated that, in evaluating the reactor internal structures for a 1.5 percent uprate condition, it used the results of the analyses performed by Westinghouse for a proposed 8 percent uprate at Waterford 3 and San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. The licensee justified the applicability of SONGS analyses on the basis of the similarity of the RVIs designs and its comparison of the hydraulic loads, fuel weight, and fuel spring loads between Waterford 3 and SONGS. The licensee determined that the hydraulic loads considered in the SONGS analyses are bounding and the current fuel weight and fuel spring loads are also bounded by the SONGS calculations. The licensee provided a summary of stresses at various reactor internal components and concluded that maximum stresses and CUFs for the reactor internal components remain within the acceptable limits of Section III of the ASME Code, 1971 Edition.

As a result of these evaluations, the licensee concluded that the reactor internal components at Waterford 3 are structurally adequate for the proposed power uprate conditions. The staff agrees with the licensee's conclusion.

3.5.1.3 Control Element Assembly Mechanisms

The licensee stated that the CEAM components are affected by the reactor coolant pressure, vessel outlet temperature, and hot leg NSSS design transients. The reactor coolant pressure (2,250 psia) and the NSSS design transients for the 1.5 percent uprate condition remain unchanged. The licensee stated that the original design basis analysis of the CEAM was evaluated for the effect of a higher vessel outlet temperature (611 ^{o}F). The effect of the 1.5 percent power uprate on vessel outlet is bounded by the original design basis temperature of 611 ^{o}F. The licensee concluded that, since there are no changes in the seismic and LOCA load conditions, the stresses and CUFs calculated for the existing design basis analysis of the CEAM components remain valid for the proposed 1.5 percent power uprate conditions.

On the basis of its review, the staff finds the licensee's conclusion that the current design of CEAMs is bounding for the power uprate conditions acceptable.

3.5.1.4 Reactor Coolant Pumps

The licensee reviewed the existing design basis analyses of the Waterford 3 RCPs to determine the impact of the revised design conditions in Table 3.3.1-1 of Reference 1. The RCS pressure for the 1.5 percent uprate condition remains unchanged. The most limiting design parameter of the SG outlet temperature, as provided in Table 3.3.1-1 of Reference 1, is decreased from 553 °F to 545 °F for the power uprate condition. Also, there are no changes to the NSSS design transients due to the power uprate. As a result of the evaluation, the licensee concluded that the existing stress analyses for the RCPs at Waterford 3 are bounding for the 1.5 percent power uprate.

On the basis of its review, the staff finds acceptable the licensee's conclusion that the RCPs, when operating at the proposed conditions with 1.5 percent power increase from the current rated power, will remain in compliance with the requirements of the codes and standards under which Waterford 3 was originally licensed.

3.5.1.5 Pressurizer

The licensee evaluated the structural adequacy of the pressurizer and components under the proposed uprated conditions. The evaluation was performed by comparing key design parameters in the current Waterford 3 pressurizer stress report against the design parameters in Table 3.3.1-1 of Reference 1 for the proposed power uprate condition. These parameters include the RCS T_{HOT}, the RCS T_{COLD}, and the pressurizer transients. The comparison shows that the changes in T_{HOT} and T_{COLD} are very small and are enveloped by the original design basis bounding parameters used in the analysis of record. The design transients are also unaffected by the uprated conditions. In addition, the pressurizer stress and fatigue analyses are not affected by the proposed power uprate condition. The licensee concluded that, for plant

operation at the 1.5 percent uprated conditions, the pressurizer components continue to meet the stress and fatigue analysis requirements of Section III of the ASME Code, 1971 Edition, including the summer 1971 Addenda, which is the Waterford 3 code of record. The staff accepts the licensee's conclusion.

3.5.1.6 Nuclear Steam Supply System Piping and Pipe Supports

The licensee evaluated the NSSS piping and supports by reviewing the design basis analysis against the uprated power condition, with regard to the design system parameters, transients and the LOCA dynamic loads. The evaluation was performed for the reactor coolant loop (RCL) piping, primary equipment nozzles, primary equipment supports, main steam, feedwater, component cooling water (CCW), and the pressurizer surge line piping systems. The methods, criteria, and requirements used in the existing design basis analysis for Waterford 3 are applicable for the power uprate evaluation.

The RCS pressure remains unchanged for the proposed core power uprate condition. In response to an item in the staff's RAI (revised Table 3.3.1-1, Reference 4), the licensee indicated that for the power uprate condition, the actual T_{HOT} is projected to be slightly greater than the T_{HOT} for the current operating condition and the T_{COLD} will be slightly less than that for the current power level condition. However, the original design basis T_{HOT} and T_{COLD} bound the T_{HOT} and T_{COLD} for the power uprate condition. The licensee stated that since Delta T (the difference between T_{HOT} and T_{COLD}) associated with the uprated condition is less than the Delta T value used in the analysis of record, the analysis of record for piping, component, and component support thermal expansion loads will remain bounding because they are associated with a higher value of Delta T.

The licensee also evaluated the LOCA condition and determined that the current LOCA hydraulic forcing functions are bounding for the uprated power condition for Waterford 3. The loads on the reactor coolant loop piping and nozzles in the existing analyses are, therefore, bounding for the proposed power uprate. The licensee indicated that the design transients used in the evaluation of the RCS piping systems and equipment nozzles are unchanged for the Waterford 3 power uprate. As a result of its review, the licensee concluded that the original design transients and analysis of record for normal operating pressure, seismic, and LOCA loads remain bounding for the power uprate condition. All piping systems remain acceptable and will continue to satisfy the design basis requirements in accordance with applicable design basis criteria, when considering the temperature pressure and flow rate effects resulting from the power uprate conditions. In Reference 4, the licensee provided primary stress intensities, primary-plus secondary stress intensity ranges, stress allowables, calculated stress margin, and the cumulative fatigue usage factors for the reactor coolant piping. Specifically, the licensee confirmed that the Waterford 3 NSSS piping and related support systems remain within allowable stress limits in accordance with ASME Section III, 1971 Edition, including summer 1971 Addenda.

On the basis of its review of the licensee's submittal, the staff finds the licensee's conclusion that the existing NSSS piping and supports, the primary equipment nozzles, the primary equipment supports, and the auxiliary lines connecting to the primary loop piping will remain in compliance with the requirements of the design basis criteria, as defined in the Waterford 3 FSAR, and acceptable for the power uprate.

3.5.1.7 Miscellaneous Systems and Components

The licensee evaluated the adequacy of the BOP systems, based on comparing the existing design bases parameters with the core power uprate conditions. The BOP piping systems evaluated for the power uprate are the main steam, feedwater, SG blowdown, feedwater heater, extraction steam, heater drains, condenser, turbine component cooling, spent fuel pool (SFP) cooling, component cooling, and circulating water systems (CWS). These piping systems, together with the RCL piping and supports, were evaluated for the effects resulting from the revised conditions in Table 3.3.1-1 of Reference 4 (RCS temperatures, steam temperature, and steam flow rate) and the heat balance at 3,441 MWt RTP. Based on its evaluation, the licensee determined that the design pressures and temperatures of the BOP piping, for the proposed 1.5 percent power uprate, remain unchanged from the original design and, therefore, existing design basis analyses for the BOP piping systems are acceptable for the uprated power level of 3,441 MWt. As a result, the licensee concluded that the Waterford 3 BOP piping and related support systems remain within allowable stress limits in accordance with ASME Section III, 1971 Edition, including the winter 1972 Addenda for Class 2, and 3 piping; and American Nuclear Standards Institute B31.1, 1973 Edition, as appropriate. The staff accepts the licensee's conclusion.

The licensee also reviewed the programs, components, structures, and non-NSSS system operational aspects affected by the power uprate. In Reference 1, the licensee stated that there are no changes to the Motor-Operated Valves program as a result of the 1.5 percent power uprate. The safety-related valves were not found to be impacted by the 1.5 percent power uprate and are, therefore, acceptable. The licensee evaluated its commitments relating to Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," that are required to operate in accordance with its intended safety function. The licensee found that the existing design parameters remain bounding for the 1.5 percent power uprate. The licensee also evaluated its response relating to GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Desin-Basis Accident Conditions," regarding the over-pressurization of isolated piping segments. The licensee concluded that the conditions used in the existing evaluation for GL 96-06 remain bounding for the proposed power level of 101.5 percent of the current power level. On the basis of the above review, the staff accepts the licensee's conclusion that the power uprate will have no adverse effects on the safety-related valves, and that the conclusions from the GL 95-07, GL 96-06, and GL 89-10 programs remain valid.

As a result of the above evaluation, the staff concludes that BOP piping, pipe supports, valves, and equipment nozzles remain acceptable and continue to satisfy the design-basis requirements for the proposed 1.5 percent power uprate.

3.5.2 Summary

On the basis of its review, the staff finds acceptable the licensee's evaluations for the NSSS and BOP piping, components, and supports; the reactor vessel and internal components; the CEAMs, SGs, RCPs; and the pressurizer at the proposed 1.5 percent power uprate. The staff finds the licensee's evaluation to be bounded by the licensing code of record and the original design basis and, therefore, concludes that the foregoing components are acceptable for the power uprate in operation of Waterford 3 at the proposed core power level of 3,441 MWt.

3.6 Balance of Plant Systems

3.6.1 Introduction

The original design of the Waterford 3 secondary side components was done for an RTP of 3,559 MWt, which bounds the proposed uprate to 3,441 MWt. Thus, the secondary side system temperatures and pressures for the 1.5 percent power uprate remain bounded by the original temperatures and pressures. The following systems were reviewed in detail for the uprated power conditions, and the review of the results are shown below.

3.6.2 Condensate and Feedwater Systems

The feedwater system supplies heated feedwater to the SGs under all load conditions, maintaining the level within the programmed band. The licensee states that for the power uprate, while the feedwater flow rate to each SG increases slightly, it still remains below the system design capabilities. Similarly, while the feedwater heater system pressure, temperature, and flow rate changes slightly at the uprated power level, these parameters still remain below the system and component design conditions. The increase in the steam flow to each condenser in the uprated power level remains bounded by the condenser design. The condensate pumps have sufficient margin to satisfy feed pump flow rate and net positive suction head at the uprated conditions.

Since these systems do not perform any safety-related function, the staff did not perform a detailed review of the impact of power uprate operations on the design and performance of these systems. However, based on the licensee's evaluation of the system and its conclusion that these systems satisfy their design bases for plant operations at the proposed uprated power level and, further, based on the experience gained from our review of power uprate applications for similar PWR plants, the staff finds that the operations of the feedwater system at the proposed uprated power level is acceptable.

3.6.3 Steam Systems

The licensee evaluated the effects resulting from plant uprated power operations on the main steam system including MSSVs, ADVs, MSIVs, main steam bypass valves, and other steam systems. The adequacy of these are addressed elsewhere in the SE.

Miscellaneous Steam Systems, such as the extraction steam system and auxilliary steam system, do not perform any safety-related function; therefore, the staff did not review the impact of plant operations under the uprated power level on the miscellaneous steam systems.

3.6.4 Main Turbine Evaluation

The capability of the Main Turbine, including the throttle valves, high-pressure and low-pressure turbines, moisture separator reheaters, and relief valves, was evaluated at the uprated power level and determined to have sufficient margin to enable operation without requiring equipment modifications. The licensee also evaluated whether there would be an increase in the probability of turbine overspeed or associated turbine missile production, due to plant power uprate operations. The licensee concluded that there would be no increase in the probability of

overspeed or associated missile production, and the turbine could continue to be operated safely at the proposed uprated power level.

Based on its review of the licensee's evaluation and the experience gained from review of power uprate applications for similar PWR plants, the staff finds that operation of the turbine at the proposed power uprate level is acceptable.

3.6.5 Circulating Water System

The CWS is designed to remove heat rejected to the condenser by turbine exhaust and other exhausts over the full range of operating loads, thereby maintaining adequately low condenser pressure. The licensee states that the total design circulating water flow rate is approximately 1,000,000 gallons per minute, which remains essentially unchanged following a power uprate, and the increased levels of rejected heat from an increase in turbine flow will increase the CWS outlet temperature to the Mississippi River by less than 0.5 °F under the power uprate conditions. The licensee concluded that operation of the CWS under the power uprate condition is bounded by the CWS system design, and additionally, the increase in the back pressure in the condenser will remain within acceptable limits. However, since the CWS does not perform any safety function, the staff has not reviewed in detail the impact of plant power uprate on the design and performance of this system.

3.6.6 Component Cooling Water and Auxiliary Component Cooling Water Systems

Cooling water for the reactor auxiliaries cooling systems is supplied by the CCW system, which in turn is cooled by the Auxiliary CCW (ACCW) system. These systems transfer the heat to the dry cooling towers or WCTs for rejection to the atmosphere, and to ensure continuous operation or safe shutdown of the unit under all modes of operations.

The licensee states that the CCW and ACCW systems will continue to remove the required heat loads for the proposed power uprate without exceeding their temperature limits, since the heat load increase due to the power uprate is bounded by the original design at 102 percent of RTP, and no changes or modifications in flow rates and operating limits are required.

Based on review of the licensee's evaluation and experience gained from the review of power uprate applications for similar PWR plants, the staff concludes that the CCW and ACCW systems are acceptable for the plant power uprate operations.

3.6.7 Spent Fuel Pool Cooling

The SFP cooling and cleanup (SFPCC) system is designed to remove the decay heat from the spent fuel assemblies stored in the SFP, and to clarify and purify the water in the SFP. The SFP cooling portion of the SFPCC system was reanalyzed for the 1998 Waterford 3 SFP Rerack Project (Reference 1), and among the several conservative assumptions used in the reanalysis were: core power of 3,661 MWt, two-year fuel cycle, 5 percent enriched fuel, etc. The licensee states that this reanalysis bounds the proposed 1.5 percent uprate and no changes are required to the SFP cooling portion of the SFPCC system.

Based on review of the licensee's evaluation and experience gained from the review of power uprate applications for similar PWR plants, the staff concludes that the SFP cooling system is acceptable for the plant power uprate operations.

3.6.8 Containment Spray System

The functions of the containment spray system (CSS) are to remove heat and fission products that may be released into the containment atmosphere following a LOCA or MSLB by spraying borated water solution into the containment, and to reduce the containment pressure.

The licensee states that the containment peak pressure and temperature analyses and the radiological consequence calculations were performed at or above 102 percent of RTP, and the 1.5 percent power uprate will remain bounded by the existing analyses.

Based on the licensee's statement and experience gained from the review of power uprate applications for similar PWR plants, the staff concludes that the CSS is acceptable for the plant power uprate operations.

3.6.9 Heating, Ventilation, and Air Conditioning Systems

The licensee evaluated the following systems to ensure that margin and capability exist to operate satisfactorily to support the plant thermal power uprate.

3.6.9.1 Containment Ventilation and Cooling Systems

The Containment Fan Cooling Subsystem is designed for use during normal and post-accident operation. During normal operation, the system provides cooling to various areas of the containment. During post-accident operation, the system removes heat from the containment atmosphere following a LOCA, MSLB, or a secondary system line break, and also maintains an acceptable containment pressure and temperature.

3.6.9.2 Control Room Heating, Ventilation, and Air Conditioning Systems

The purpose of the Control Room HVAC system is to maintain Control Room envelope, which includes the Control Room, computer room, etc., in a habitable condition, to assure that the operators can remain in the Control Room to operate the plant safely during normal conditions, and maintain the unit in a safe condition under design basis accident conditions or in the event of an FSAR-analyzed offsite toxic chemical accident.

3.6.9.3 Reactor Auxiliary Building Heating, Ventilation, and Air Conditioning System

The purpose of the Reactor Auxiliary Building (RAB) Normal Ventilation System is to cool and heat parts of the RAB during normal operation, to purge the Reactor Containment when operating in the normal combined with containment purge mode, or to provide ventilation in the normal combined with refueling ventilation mode.

The licensee states that the containment peak pressure and containment peak temperature analyses, radiological consequence calculations, original design bases for the onsite

radiological releases, and heat loads due to accident conditions were performed at or above 102 percent of RTP, and the 1.5 percent power uprate will remain bounded by the existing analyses.

Based on review of the licensee's evaluation and experience gained from the review of power uprate applications for similar PWR plants, the staff concludes that the Containment Ventilation and Cooling System is acceptable for the plant power uprate operations.

3.6.10 Radioactive Waste Management

Radioactive wastes are processed through the Solid, Liquid, and/or Gaseous Waste Management System. The licensee states that the original design of these systems was based on an RTP of 3,560 MWt, which bounds the proposed power uprate to an RTP of 3,441 MWt, since the fuel design, the maximum burnup, and probability of fuel failure are all unaffected by the uprate.

Based on this statement and experience gained from the review of power uprate applications for similar PWR plants, the staff concludes that the Solid, Liquid, and Gaseous Waste Management Systems are acceptable for the plant power uprate operations.

3.6.11 Containment Accident Analysis

The licensee reviewed the existing containment integrity analysis to ensure the maximum pressure inside the containment would not exceed the containment design pressure if a design basis LOCA or MSLB inside containment should occur during plant operation. The review also established the pressure and temperature for environmental qualification and operation of safety-related equipment located inside the containment. The results of the review follow.

3.6.11.1 Containment Integrity Analysis - Loss-of-Coolant Accident

The licensee states in Section 3.11.1.2 and Table 3.11.1.2-1 of Reference 1 that the current mass and energy release data for input into the containment response analysis were generated at an RTP of 3,734 MWt, which is based on 108 percent of the current license power level of 3,390 MWt, plus an additional 2 percent that accounts for measurement inaccuracy. This data bounds the power uprate levels. Therefore, the mass and energy release data for the LOCA bound the power uprate conditions, and the peak LOCA containment pressure and temperature will not be affected by the power uprate. The capability of the containment heat removal system to reduce the containment pressure by one-half of the peak pressure within 24 hours following a LOCA is also unaffected by the power uprate.

3.6.11.2 Containment Integrity Analysis - Main Steam Line Break

The licensee states in Section 3.11.1.3 and Table 3.11.1.3-1 of Reference 1 that the 1.5 percent power uprate has the potential to affect the mass and energy released to the containment during a steam line break. The existing analysis for the limiting mass and energy release due to a MSLB was performed from a core power of 3,457.8 MWt, and the results remain applicable for the 1.5 percent power uprate.

Based on review and assessment of the information provided in the licensee's submittal, the staff finds acceptable the licensee's conclusion that the peak LOCA, as well as MSLB containment pressure and temperature, will not be affected by the power uprate, and the containment integrity analysis at the proposed uprated power is bounded by current LOCA and MSLB analysis.

3.6.12 Summary

Based on its review of the licensee's rationale and evaluation, its evaluations described above, and experience gained from review of power uprate applications for similar PWR plants, the staff concludes that Waterford 3 operations at the proposed power uprate is acceptable.

3.7 Electrical Power

3.7.1 Introduction

Operating at the proposed increased thermal power rating will result in a corresponding increase in the electrical output of Waterford 3. The staff has reviewed the impact of the proposed increase on the plant electrical power systems (offsite, onsite, and direct current (dc)), on the SBO coping capability, and on the environmental qualification (EQ) of electrical equipment.

The staff requested the licensee to provide additional information on certain items in the licensee’s original application (Reference 1) and, by Reference 2, the licensee provided the information.

3.7.2 Evaluation

3.7.2.1 Offsite Power

The staff has reviewed information provided by the licensee to determine the impact of the power uprate on offsite power. The areas reviewed were the grid stability analysis and related electrical systems.

3.7.2.1.1 Grid Stability and Reliability Analysis

The licensee stated that grid analysis was performed for a bounding uprate of 2 percent, assuming a bounding gross generator output of 1,172 (1,150 +2 percent) MW. The analysis resulted in the conclusion that there is no impact on grid stability and reliability for a power uprate of 1.5 percent. Additionally, the Waterford 3 power uprate will not adversely impact the availability of the offsite power source for Waterford 3 house loads in the event of a unit trip. Based on review of the current analysis, the licensee determined that current grid stability and reliability are not impacted, and Waterford 3 continues to be in conformance with 10 CFR Part 50, General Design Criterion (GDC)-17, "Electric Power Systems," for the power uprated electrical conditions.

In response to the staff’s RAI (Reference 2), the licensee stated that the grid stability analysis assumed projected load growth and a 2 percent power increase in electrical output of

Waterford 3. The analysis concluded that the grid and Waterford 3 generator will remain stable for postulated faults in the vicinity of Waterford 3.

On the basis of this information, the staff concludes that the proposed power uprate to 3,441 MWt at Waterford 3 will not adversely affect the grid stability and reliability. Therefore, the staff has reasonable assurance that GDC-17 will be met for all power uprate conditions.

3.7.2.2 Related Electrical Systems

The licensee performed a power uprate review to determine the adequacy of electrical systems associated with the main turbine-generator auxiliary systems. The staff review follows.

3.7.2.2.1 Main Generator

The existing main generator is rated at 1,333.2 MegaVolt Amperes (MVA) and 25 kilo Volts (kV). The expected generator output is 1,200 MW at 0.9 lagging power factor when operating with 60 psig hydrogen pressure. The net plant power uprate-related output will be 1,120.5 MW, and is within the capability of the generator. The staff's review determined that the electrical system's configuration and operating voltage ranges would be unchanged and would remain adequate for operation at the higher output.

3.7.2.2.2 Isolated Phase Bus Duct

To deliver electrical power from the generator to the transmission system, the unit is equipped with a main isolated phase (isophase) bus and split into two secondary isophase buses; one for each of the two main transformers, cabling, and two switching station breakers. All components are rated to deliver electrical power at or in excess of the main generator rating of 1,333.2 MVA.

The existing isophase bus duct rating is 33,000 amperes (amps) and 25 kV for the main section. The bus conductor is rated for a temperature of 65 °C rise with forced cooling. The maximum current output is 32,409 amps [1,333,200/(1.7321x25x0.95)], with a main generator output of 1,333.2 MVA and 95 percent of 25 kV. Each of the secondary isophase buses are rated at 15,000 amps at 65 °C rise with forced cooling, and has an increase rating of 20,000 amps at 65 °C rise with emergency cooling. The review determined that the isophase bus duct would be adequate for both rated voltage and low-voltage current output for the power uprate. The staff determined that the isophase bus duct would be adequate for a power uprate of 3,441 MWt.

3.7.2.2.3 Main Transformers

There are two main transformers at Waterford 3. The main transformer rating is 600 MVA and 25/161 kV. The main transformers are of the forced oil and forced air cooled type. When operating with both cooling systems on a single transformer, the rating of the transformer still

operating can be increased from 600 MVA to 798 MVA for the same rated temperature rise of 65 °C. Therefore, the main transformers will support the power increase with no modifications.

3.7.2.2.4 Switchyard Equipment

The licensee stated that switchyard equipment are designed to meet or exceed the rated capacity of the main generator per standard design practice at Entergy. Waterford 3 switchyard equipment will accept the additional MW load without the need for any hardware modifications.

Thus, the turbine-generator and major electrical components from the isophase bus to the switchyard have adequate margin to accept the additional power anticipated by the 1.5 percent uprate and GDC-17 will continue to be met.

3.7.2.3 Onsite Power

The onsite ac power system includes a Class 1E system and a non-Class 1E system. The onsite ac power system consists of the Unit 3 main turbine-generator, two unit auxiliary transformers, two EDGs, and an ac distribution system with nominal ratings of 6.9 kV, 4.16 kV, 480 volts (V), and 208/120 V. The onsite dc system, consisting of Class 1E and non-Class 1E systems, provides control power for medium-voltage and low-voltage switchgear, diesel generator controls, and other control systems.

The 1.5 percent power uprate does not result in higher loading of any pumps or other mechanical equipment. Hence, motor loading is not affected. The slightly higher heat input in the primary and secondary systems will result in a small increase in the duration of equipment operation, but does not impact the continuous rating of electrical equipment. Hence, the electrical loading of plant equipment is not impacted and no changes are anticipated.

3.7.2.3.1 Non-Class 1E Alternating Current System

The non-Class 1E ac system distributes power at 6.9 kV, 4.16 kV, 480 V, and 208/120 V for all non-safety-related loads. The non-Class 1E ac buses normally are supplied through the unit auxiliary transformers from the main generator. However, during plant startup, shutdown, and post-shutdown, power is supplied from a 230 kV preferred offsite power source through the secondaries of the startup transformers consisting of dual windings (230kV / 6.9 kV, 4.16 kV).

The 4.16 kV non-Class 1E auxiliary system is comprised of four buses. The large non-safety-related loads fed from these buses include heater drain pumps, non-safety chillers, and turbine cooling water pumps. The majority of loads supplied from these buses are at the 480 V level. The 6.9 kV buses power the circulating water pumps and condensate pumps for the secondary system. The secondary side was originally designed for a NSSS rating of 3,559 MWt. Hence, the large pumps and motors on the 6.9 kV and 4.16 kV buses are adequately sized. The cables and protective relaying are based on nominal rating of the motors, and these are not affected.

The RCPs are fed from non-Class 1E 6.9 kV auxiliary system buses. The 1.5 percent increase in thermal power does not affect the T_{COLD} and, therefore, the RCPs loading is not affected, nor are the cables and protective relaying for the RCPs. The non-Class 1E startup transformers

are capable of supplying all of the startup or normal plant operating loads of the unit or the ESF loads. The 1.5 percent power uprate will not increase the electrical loading of the transformers. Hence, the existing ratings of the transformers will be adequate.

3.7.2.3.2 Class 1E Alternating Current System

The Class 1E ac system consists of two separate trains and distributes power at 4.16 kV, 480 V, and 120 V to safety-related loads. The Class 1E ac buses are normally supplied through the unit auxiliary transformers from the main generator. The 4.16 kV Class 1E auxiliary system is comprised of two buses. A swing bus is available to replace either bus for maintenance.

Each safety-related 4.16 kV bus is supplied by offsite power through the startup transformer and one standby EDG. In the event of a loss-of-offsite power (LOOP), the Class 1E ac system will be powered from the EDGs.

The large 4.16 kV loads consist of high-pressure safety injection, low-pressure safety injection, containment spray, EFW, CCW, and ACCW pumps. The power uprate of 1.5 percent is within the design bases of the original design of the plant for operation at 102 percent plant rating. The large pumps and motors are sized for safe shutdown following a design basis event, with the plant at an initial power of 102 percent. The electrical motors are sized for maximum pump loading requirements. The cables and protective relaying are based on the nominal rating of the motors plus design margins. Hence, the proposed change in plant power does not require uprating the existing pumps, cables, or motors. The continuous and short circuit ratings of the switchgear are not affected by the small change in plant power.

In response to the staff's concern about the equipment terminal voltages due to power uprate, the licensee, by Reference 2, stated that the terminal voltage for safety-related and non-safety-related equipment is governed by degraded grid conditions that are monitored by degraded voltage (DV) relays. The electrical equipment has adequate terminal voltage at the trip point of the DV relays. Since there is no significant increase in the plant electrical loads and the grid system is not adversely impacted due to this power uprate (or other generation changes), there is no change required in the DV relay setpoint. The setting of these relays (93.1 percent of 4,160 V ac (Vac)) is based on a conservative loading of plant auxiliaries to yield the worst case voltage drop for the electrical system.

The existing EDGs are rated at 4.4 MW with 10 percent overload capability. The maximum calculated accident loading is expected to be approximately 4.2 MW. Although the electrical loading is not expected to change, there is adequate margin in the normal rating of the EDGs to accommodate any minor variations in electrical loads. The minor increase in post accident decay heat load is considered within the error margin for the time-based fuel oil consumption calculation.

3.7.2.3.3 120 Volt Alternating Current and 125 Volt Direct Current Systems

The dc system is made up of four trains, each of which has a battery, two battery chargers, and power distribution panels. The chargers convert 480 Vac to dc using silicon controlled rectifiers and silicon diodes. The major dc loads on the Class 1E systems are the static un-interruptible

power supplies (SUPS) which power the 120 Vac system and control power for switchgear and critical valves. The system consists of nine SUPS, of which six are safety-related and three are non-safety-related. The change in power requirements for the six safety-related SUPS due to power uprate is insignificant. The minor change in non-safety-related SUPS loading due to additional power requirements of the proposed use of the Caldon LEFM ✓+™ instrumentation has been evaluated to be acceptable for the respective panels.

The major loads on the Class 1E dc system powered by the turbine generator building battery are emergency lube oil and seal pumps. These pumps are required for turbine coastdown upon reactor trip and LOOP event. The turbine speed is not affected by the 1.5 percent power uprate. Since the coastdown time is a direct function of the turbine momentum and turbine mass and speed are not affected, there is no impact on the coastdown time and battery loading.

The low-voltage distribution and lighting (LVD) system supplies 208 Vac and 120 Vac power to various plant loads, both safety-related and non-safety-related, and provides various types of lighting to all areas of the plant. The LVD system is comprised of a safety-related power distribution system, a non-safety-related power distribution system, and a lighting power distribution system. The LVD system is physically connected to virtually every system in the plant. It provides power for numerous uses, such as motor space heaters, solenoid valves, relays, ventilation dampers, lighting, controls and indications, annunciators, etc. There are no changes to the loads at the 120 Vac and 208 Vac system.

On the basis of its review, the staff finds that the electrical equipment powered by the onsite distribution system remains within its respective ratings. The staff has reasonable assurance that Waterford 3 will continue to meet GDC-17 for the power uprate of 3,441 MWt.

3.7.2.4 Station Blackout

The staff has reviewed information provided by the licensee to determine the impact of the power uprate on the existing analysis for a SBO. The licensee reevaluated its SBO analysis using the guidelines of Nuclear Management and Resources Council (NUMARC) 8700, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," except where RG 1.155, "Station Blackout," takes precedence. The licensee stated that the plant responses to and coping capabilities for an SBO event would be affected slightly by operation at the power uprate RTP level, due to the increase in decay heat. There are no changes to the systems or equipment used to respond to SBO, nor is the required coping time changed.

The licensee stated that there is a slight increase in decay heat generation (slightly higher cooling load during cooldown) for the proposed uprate from 100 percent to 101.5 percent (3,390 MWt to 3,441 MWt). However, the containment pressure and temperature profiles will continue to be bounded by the existing LOCA profiles.

The necessary condensate inventory required for decay heat removal for 100 percent power with 20 MWt of RCP's decay heat is calculated to be 75,429 gallons. The new condensate inventory required for decay heat removal, as a result of the proposed change (3,441 MWt with 20 MWt of RCP's decay heat) is 76,557 gallons. Both of these quantities are less than the TS

minimum requirement of 170,000 gallons for the CSP, thus, the plant's current condensate inventory is adequate.

The ADVs were designed to provide a means of decay heat removal and plant cooldown during loss of condenser vacuum from a steady state power of 100 percent RTP+2 percent instrument uncertainty. This design bounds the power uprate.

Other elements of the SBO analysis have not significantly changed: plant lighting, RCS inventory, shutdown margin, containment isolation, loss of ventilation, compressed air, battery capacity, coping period, diesel generator reliability, or equipment required operable for a SBO. None of the SBO-associated instruments require control setpoint changes, and none of the associated instruments exceed the design basis due to the power uprate. Therefore, the SBO analysis is not affected by this power uprate.

In response to the staff's concern about the battery margin, the licensee, in Reference 2, stated that the dc loads and duration of the event remain the same with the power uprate. Hence, there is no adverse impact on the battery capacity calculation, and the battery margins remain the same.

On the basis of its review, the staff concludes that the power uprate conditions do not result in any significant changes to the previous SBO evaluation and, therefore, Waterford 3 will continue to meet 10 CFR 50.63 requirements.

3.7.2.5 Environmental Qualification

3.7.2.5.1 Electrical Equipment

The licensee stated that current containment LOCA and MSLB analyses will not be affected by uprate conditions.

3.7.2.5.2 Inside Containment

The licensee stated that the current EQ accident environments inside containment bound the environments resulting from the power uprate. The current evaluations performed per 10 CFR 50.49, therefore, remain valid.

3.7.2.5.3 Outside Containment

In response to the staff's RAI, the licensee, in Reference 2, stated that containment accident parameters remain bounded to the environments resulting from the power uprate. As a result, the safety-related equipment located outside containment will also not be subjected to accident environments more severe than those postulated for current design basis conditions. The current evaluations performed per 10 CFR 50.49, therefore, remain valid.

On basis of its review, the staff finds that the power uprate has a negligible affect on the environmental conditions currently used by the EQ program for safety-related electrical equipment located both inside and outside the primary containment. The staff concludes that Waterford 3 will continue to meet 10 CFR 50.49 requirements.

3.7.3 Summary

The staff has reviewed the information in Refrences 1 and 2 and concludes that the proposed power uprate does not adversely impact the plant electrical power systems (offsite, onsite, and dc), the SBO coping capability, or the environmental qualification of electrical equipment. This is consistent with GDC-17, 10 CFR 50.63, and 10 CFR 50.49; therefore, the proposed power uprate is acceptable. In addition, the staff finds that this amendment does not require any TS changes for the electrical power systems.

3.8 Instrumentation and Control Systems

3.8.1 Introduction

References 5 and 7 provide the generic basis for increasing power by as much as 1.5 percent. The staff approved References 5 and 7 in SERs dated March 8, 1999, and November 16, 2001, respectively. The licensee's submittal provides a plant-specific justification for the proposed 1.5 percent power uprate at Waterford 3 on the basis of References 5 and 7. This SER addresses the licensee's plant-specific justification for a 1.5 percent power uprate.

3.8.2 Background

The instrumentation for measuring feedwater flow rate typically uses a venturi, an orifice plate, or a flow nozzle to generate a differential pressure proportional to the feedwater velocity in the pipe. The most common instrumentation for measuring flow rates is the venturi flow meter in the feedwater system piping, such as at Waterford 3. The major advantage of a venturi flow meter over the other two flow measurement designs is the relatively low head loss created as the feedwater passes through the device. The major disadvantage of the venturi flow meter is the effect of venturi fouling upon flow meter instrument accuracy. Fouling causes a venturi flow meter to indicate higher differential pressures for equivalent flow velocities, which results in an output signal representing a higher than actual flow rate. Since feedwater flow rate is directly proportional to calorimetric power, this error in feedwater flow rate measurement leads the plant operator to calibrate the nuclear instrumentation at a higher than actual core power.

Calibrating the nuclear instrumentation to indicate higher than actual core power is conservative with respect to reactor safety, but causes the licensee to generate electrical power proportionately lower when the plant is operated at its indicated thermal power rating. To eliminate the effects of venturi fouling, the venturi flow meter device must be removed, cleaned, and calibrated. The high cost of flow meter calibration and the need to improve flow instrumentation accuracy prompted the nuclear industry to assess other flow measurement

techniques. Use of a LEFM-implementing transit time technology was found to be a viable alternative.

The basis of using transit time technology to measure fluid velocity is that ultrasonic pulses transmitted into a fluid stream travel faster in the direction of the fluid flow than opposite the flow. Consequently, the difference in the upstream and downstream traversing times is proportional to the velocity of the fluid in the pipe that has been traversed by the ultrasonic pulses. (Additionally, the average of the upstream and downstream transit times is proportional to the average density of the traversed fluid, which is a function of the average fluid temperature and pressure.)

The Caldon Chordal LEFM ✓+™ is a digital system controlled by software that consists of an electronic cabinet in the auxiliary instrument room and a measurement section, or a spool piece, permanently mounted in each of the feedwater pipes. The LEFM ✓+™ measures eight line integral velocities at precise locations with respect to the pipe center line. The system numerically integrates the eight measured velocities to determine the average (bulk) feedwater flow rate and the bulk feedwater fluid temperature. These processed measurements are then used by the plant computer to determine the reactor thermal power.

The staff's SE of the licensee's submittal is discussed in the following section.

3.8.3 Evaluation

References 5 and 7 (both approved by the staff) describe the LEFM ✓+™ system for the measurement of feedwater flow and temperature to determine reactor thermal power. The LEFM ✓+™ system provides a basis for a 1.5 percent uprate of the licensed reactor power. References 5 and 7 state that the LEFM is superior to the venturi-based instrumentation currently in use on the basis of the following:

1. The elements of LEFM ✓+™ accuracy can be verified on-line,

2. The LEFM ✓+™ measurement accuracy results in a uncertainty ±0.5 percent of thermal power, with a 95 percent confidence limit, whereas the measurement uncertainty of the current venturi flow element instrumentation is ±1.4 percent.

3. The licensee used an approved setpoint methodology to calculate the plant-specific total power measurement uncertainty of the LEFM ✓+™. The calculation was done with two standard deviations to determine with a 95 percent confidence level (probability of operation within bounds) that the calculated power measurement uncertainty bounds of the LEFM ✓+™ was less than ±0.5 percent.

In approving References 5 and 7, the staff required that licensees submit, as part of the power uprate package, a detailed accounting of the uncertainties applicable to the licensed facility, and additionally, that licensees maintain, as part of the plant design documents, the LEFM calibration and other data justifying the proposed power uprate. The licensee provided the requested uncertainty information, with clarifications regarding the initially declared (1-σ) uncertainty limits instead of the required (2-σ) uncertainty limits required for a 95 percent confidence interval. On the basis of the staff's review of the uncertainty values

associated with performing a secondary power calorimetric calculation, the staff finds the (2-σ) uncertainty values acceptable for the requested 1.5 percent power uprate.

The licensee stated that the implementing modification package specifies the affected maintenance and operating procedures that must be in place prior to declaring the units operable and raising the power above 3,390 MWt. Additionally, the licensee stated the system software had been developed and will be maintained under a validation and verification (V&V) program. The V&V program has been applied to all system software and includes a detailed code review. The staff finds these actions address the staff's quality assurance records maintenance requirements for implementing the LEFM in an acceptable manner.

Additionally, as part of approving References 5 and 7, the staff included four requirements to be addressed by a licensee requesting a power uprate. The licensee addressed each of these four requirements as follows:

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

The licensee stated that implementation of the power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level with the new LEFM ✓+™ system. Additionally, the licensee stated, plant maintenance and calibration procedures will be revised to incorporate Caldon's maintenance and calibration requirements prior to declaring the LEFM ✓+™ system operable and raising power above 3,390 MWt. The incorporation of and continued adherence to these requirements will assure that the LEFM ✓+™ system is properly maintained and calibrated.

The licensee further stated that the LEFM ✓+™ operability requirements will be contained in the Waterford 3 Technical Requirements Manuals (TRM). An LCO has been drafted for inclusion in the TRM stating that an operable LEFM ✓+™ shall be used in the performance of the calorimetric heat balance measurements whenever power is greater than the pre-uprate level of 3,390 MWt. If the LEFM ✓+™ is not operable, plant operation will be administratively controlled at a power level consistent with the accuracy of the available instrumentation. With these controls, the effect on plant operations is that power will be maintained at a level that accounts for the appropriate instrumentation uncertainties, thereby preserving ECCS limits. The staff finds these actions consistent with this criterion and, therefore, acceptable.

Requirement 2. For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed instrumentation, and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Reference 5.

The licensee stated that this criterion is not applicable to Waterford 3 because the plant currently uses venturi flow meters to obtain the calorimetric heat balance measurements. The licensee is installing a new LEFM ✓+™ during Refueling Outage Eleven as the basis for the requested uprate. The staff finds the licensee's response to this criterion acceptable.

Requirement 3. Confirm that the methodology used to calculate the uncertainty of the LEFM, in comparison to the current feedwater instrumentation, is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

The licensee stated that the uncertainty associated with the LEFM ✓+™ and the method used to derive that uncertainty is described in Reference 7. The analysis described in that report was performed to determine and confirm the total secondary calorimetric power measurement uncertainty based on using the LEFM ✓+™ flow meters as preferred inputs and as calibration inputs to the existing feedwater and main steam venturi flow instrumentation loops. This analysis, the licensee stated, compares the uncertainties of the existing flow measurement system to the LEFM ✓+™ units. The staff reviewed the analysis described in Reference 7 and found it acceptable, as described in its safety analysis (Reference 8). The staff, therefore, finds the licensee has addressed this criterion in an acceptable manner.

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

The licensee stated that this criterion does not apply to Waterford 3. The calibration factors for the Waterford 3 spool pieces were established by tests of these spools at Alden Research Laboratory (Alden) in June 2001. These tests included a full scale model of the Waterford 3 hydraulic geometry and tests in a straight pipe. An Alden data report for these tests and a Caldon engineering report evaluating the test data will be maintained on file by the licensee. These reports provide the bases for the calibration factor used for the LEFM ✓+™ at Waterford 3. The Caldon engineering report provides the bases for the uncertainty in the calibration factor for the spools. The site-specific uncertainty analysis will document these analyses. This document will be maintained on file, as part of the technical basis for the Waterford 3 uprate.

The licensee further stated that final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process. The commissioning process verifies bounding calibration test data (See Appendix F of References 5 and 7). This step provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation, as described in the licensee's request for a power uprate. Final commissioning is expected to be completed in April 2002. The staff finds that these actions address the purpose of this criterion and, therefore, are acceptable.

The staff finds that the licensee response to the above four criteria has acceptably resolved the plant-specific concerns about LEFM ✓+™ maintenance and calibration, hydraulic configuration, processes, and contingencies for an inoperable LEFM ✓+™, and the methodology for the plant-specific calculations of the LEFM ✓+™ power measurement uncertainty.

3.8.4 Summary

The staff evaluation found the calculation of the power calorimetric measurement uncertainty for the Waterford 3 power uprate to be acceptable. On the basis of the staff's review of the licensee plant-specific LEFM ✓+™ error band calculation, the staff finds that the Waterford 3 LEFM ✓+™ thermal power measurement uncertainty is less than 0.5 percent of actual reactor thermal power and can support the proposed 1.5 percent uprate of the Waterford 3 licensed thermal power. The staff also found that the licensee sufficiently addressed the four additional criteria outlined in the staff SER of References 5 and 7. The staff, therefore, finds the licensee request for a 1.5 percent thermal power uprate to be acceptable.

3.9 Human Factors

3.9.1 Background

The NRC staff reviewed the operator performance aspects of the Waterford 3 submittal (Reference 1) and Reference 3, which provided a response to the staff's RAI. The staff's evaluation of the licensee's request for a license amendment to permit a power uprate, as it relates to operator licensing and human performance based on five review topic areas, is as follows.

3.9.2 Evaluation

3.9.2.1 Changes in Emergency and Abnormal Operating Procedures

In Reference 1, the licensee stated that:

> "The power uprate is expected to have a limited affect on the manner in which the operators control the plant, either during normal operations, transient or emergency conditions. The power uprate will lead to minor changes in several plant parameters which include the 100 [percent] value for Rated Thermal Power, 100 [percent] Licensed Power Limits, Reactor Coolant system delta temperature, 100 [percent] Turbine Governor Valve Position, New Power Operating Limits for LPD and DNBR, Main Turbine Impulse Pressure, Steam Generator Pressure, and Main Feed Water and Steam Flows. Changes associated with the power uprate will be treated in a manner consistent with any other plant modification. In addition, the COLSS Licensed Power Monitoring algorithm will be modified and this will be identified and included in the [Section entitled Operator Training and Simulator]. The Waterford 3 Technical Requirements Manual will be revised for the LEFM CheckPlus out of service power reduction described in Section 3.2."

In the supplemental response (Reference 3), the licensee indicated that:

> "The Waterford 3 change control process requires the identification and update of the affected operating procedures associated with a modification. The procedures that impact plant operation will be revised prior to operation above the current licensed power level."

The staff finds that the licensee's response is satisfactory because the licensee has adequately identified the type and scope of plant procedures that will be affected by the uprate, and indicated that the procedures will be appropriately revised and operators will be trained on the changes before the procedures are implemented.

3.9.2.2 Changes to Risk-Important Operator Actions Sensitive to Power Uprate

In Reference 1, the licensee stated that:

> "The Waterford 3 Probabilistic Risk Assessment (PRA) model is a Level 2 analysis which includes both core damage frequency and containment performance. The success criteria used were derived from both FSAR and best estimate analyses. The Appendix K power uprate of 1.5 percent will have a negligible impact on these success criteria analyses. Timing for events and human actions will not be significantly impacted for this small increase in core power."

The staff finds the response to be satisfactory.

3.9.2.3 Changes to Control Room Controls, Displays, and Alarms

In Reference 1, the licensee stated that:

> "A Control Room alarm will be added due to the installation of the LEFM CheckPlus System. This alarm will be added to the appropriate Alarm Response Procedure (ARP) as described in the design change package which implements the installation of this new equipment. This ARP will specify actions required upon loss of the LEFM CheckPlus instrument, including entry into the TRM Action required when this new instrumentation is not functioning properly.
>
> Control Room indicators for Reactor Power will display 100 [percent] power for the new 3,441 MWt power level. Other plant parameters will have minor changes. Those parameters determined to be outside of their existing indicating bands will be addressed within the design change package which implements all of the additional plant changes (including span and scaling changes) due to this power uprate other than the installation of the LEFM CheckPlus System."

In the supplemental response (Reference 3), the licensee indicated that:

> "... The alarm response procedure will be updated prior to startup from the outage to reflect the new alarm. Thus, these changes will be available prior to operation above the current licensed power level."

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

3.9.2.4 Changes in Safety Parameter Display System

In Reference 3, the licensee stated that:

> "The power uprate will have [a] negligible impact on the Waterford 3 Safety Parameter Display System (SPDS). All points will remain within their existing ranges. Affected operating values, such as RCS temperature, Steam Generator pressures, and associated flows are addressed within applicable operating procedures."

The licensee stated that operation under uprated power conditions will have a negligible effect on the parameter displays of the SPDS. The staff finds the licensee's response to be satisfactory.

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

In Reference 1, the licensee stated:

> "Physical changes (hardware) that affect the control room and the simulator will be implemented through plant approved change processes....The necessary procedures and training documents required for operation at the uprated power level with the new LEFM CheckPlus System will be identified in the design modification package....Classroom and Simulator training will be provided on all changes that affect operator performance caused by this power uprate..."

In Reference 3, the licensee also states that:

> "...[T]he current implementation schedule for the simulator changes to reflect the power uprate modification includes a completion date prior to operation above the current licensed power level.... [and] will be complete prior to operation at elevated power levels....
>
> Based on the current plans for implementation of COLSS program changes and the update to the simulator PMC system, these changes to the simulator will be implemented prior to operation at power levels above the plant's current licensed power level."

The staff finds the licensee’s response satisfactory because it has adequately described how the changes to operator actions will be addressed by the simulator and how the simulator will accommodate the changes. Additionally, the licensee's provisions and controls for implementing the changes to the Operator Training Program and the Control Room Simulator are satisfactory and acceptable.

3.9.3 Summary

The NRC staff finds that the licensee has adequately identified changes resulting from the power uprate on control room alarms, displays, and controls; emergency and abnormal operating procedures; the SPDS; and the operator training program and the control room simulator. In those instances where changes were identified, the licensee has indicated that appropriate actions would be taken to ensure that operator performance would not be adversely affected by operation at the uprated power level. Specifically, the licensee has proposed measures acceptable to the NRC staff to incorporate changes associated with the proposed

power uprate to control room displays, alarms, and operator training and control room simulation.

4.0 STATE CONSULTATION

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

5.0 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 and there has been no public comment on such finding (66 FR 55017, published October 31, 2001). 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.

6.0 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.

7.0 REFERENCES

1. Letter from John T. Herron (Entergy) to NRC Document Control Desk, “Waterford 3 SES [Steam Electric Station], Docket No. 50-382, License No. NPF-38, Technical Specification Change Request, NPF-38-238, Appendix K Margin Recovery - Power Uprate Request,” September 21, 2001.

2. Letter from Alan J. Harris (Entergy) to NRC Document Control Desk, “Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38, Technical Specification Change Request, NPF-38-238, Appendix K Margin Recovery - Power Uprate Request, Response to Requests for Additional Information,” December 10, 2001.

3. Letter from Barry S. Allen (Entergy) to NRC Document Control Desk, “Waterford Steam Electric Station, Unit 3, Docket No. 50-382, Technical Specification Change Request, NPF-38-238, Appendix K Margin Recovery - Power Uprate Request, Response to Requests for Additional Information,” January 16, 2002.

4. Letter from John T. Herron (Entergy) to NRC Document Control Desk, “Waterford Steam Electric Station, Unit 3, Docket No. 50-382, Technical Specification Change Request, NPF-38-238, Appendix K Margin Recovery - Power Uprate Request, Response to Request for Additional Information,” January 21, 2002.

5. Caldon ER-80P, Revision 0, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM™ System,” Caldon Incorporated, March 1997.

6. J. N. Hannon (NRC) letter to C. L. Terry (TU Electric), “Staff Acceptance of Caldon Topical Report ER-80P: Improving Thermal Power Accuracy While Increasing Power Level Using the LEFM System,” March 8, 1999.

7. Caldon ER-157P, “Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM™ Check or LEFM CheckPlus™ System,” Revision 5, Caldon Incorporated, October 2001.

8. S. A. Richards (NRC) letter to M. A. Krupa (Entergy), “Waterford Steam Electric Station, Unit 3; River Bend Station; and Grand Gulf Nuclear Station - Review of Caldon, Inc. Engineering Report ER-157P,” December 20, 2001.

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

10. Letter from B. S. Allen, Entergy, to NRC, “Waterford Steam Electric Station, Unit 3, Docket No. 50-382, Supplement to Amendment Request NPF-38-234, Replacement of Part-Length Control Element Assemblies," Letter No. W3F1-2002-0004, January 17, 2002.

11. Letter from J. T. Herron, Entergy, to NRC, “Waterford Steam Electric Station, Unit 3, Docket No. 50-382, Supplement to Amendment Request NPF-38-234, Replacement of Part-Length Control Element Assemblies," Letter No. W3F1-2002-0013, February 1, 2002.

12. Letter from B. S. Allen, Entergy, to NRC, “Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License Amendment Request NPF-38-241, Revision to Peak Linear Heat Rate Safety Limit Technical Specification 2.1.1.2," Letter No. W3F1-2002-0012, January 31, 2002.

13. Letter from R. P. Barkhurst, Entergy, to NRC “Technical Specification Change Request NPF-38-148," dated December 14, 1993.

14. Letter from R. F. Burski, Entergy, to NRC “Request [for] Additional Information Regarding Technical Specification Change Request NPF-38-148," dated March 3, 1995.

15. Letter from C. M. Dugger, Entergy, to NRC “Technical Specification Change Request NPF-38-218, Extend Pressure Temperature Curve to 20 EFPY,” dated July 15, 1999.

16. Letter from C. M. Dugger, Entergy, to NRC “Technical Specification Change Request NPF-38-218, Revision 1, Extend Pressure Temperature Curve to 16 EFPY,” dated January 6, 2000.

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