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PR

September 21, 2001

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Technical Specification Change Request, NPF-38-238
Appendix K Margin Recovery – Power Uprate Request

Gentlemen:

In accordance with 10CFR50.90, Entergy Operations, Inc. (Entergy) is hereby requesting approval of changes to the Waterford 3 Operating License and Technical Specifications associated with an increase in the licensed power level. The changes involve a proposed increase in the power level from 3,390 MWt to 3,441 MWt. These changes result from increased feedwater flow measurement accuracy to be achieved by utilizing high accuracy ultrasonic flow measurement instrumentation. The proposed changes are described in Attachment 1.

Entergy has endeavored to propose only those license and Technical Specification (TS) changes that are required in order to implement the increased power level.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

Entergy requests that the effective date for this TS change to be within 60 days of startup from Refueling Outage (RF) 11. Although this request is neither exigent nor emergency, your prompt review and approval prior to startup from RF 11 is requested. Entergy would like to implement the increased power

A001

level upon startup from our upcoming RF11 scheduled to start on March 22, 2002.

Entergy notes that various Combustion Engineering topical reports that are a part of the Waterford 3 licensing basis (e.g., CENPD-132P, Calculative Methods for the C-E Large Break LOCA Evaluation Model) may have included explicit references to their use of "102% of licensed core power levels." Entergy does not consider that these topical reports require revision to reflect this requested power uprate. Rather, it will be understood that those statements refer to the Appendix K margin and the original licensed power level.

A summary of the commitments associated with the implementation of this request is provided in Attachment 4. Should you have any questions or comments concerning this request, please contact Jerry Burford at (601) 368-5755.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 21, 2001.

Very truly yours,

J. T. Herron
Vice President, Operations
Waterford 3

JTH/FGB/cbh

Attachments

cc: E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office
Louisiana DEQ/Surveillance Division
American Nuclear Insurers

ATTACHMENT 1

TO

W3F1-2001-0091

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

DESCRIPTION

Entergy Operations, Inc. (Entergy) is proposing that the Waterford 3 Operating License be amended to reflect an increase in the licensed reactor power level from 3,390 MWt to 3,441 MWt (an approximate 1.5% increase). These changes result from increased feedwater flow measurement accuracy to be achieved by utilizing high accuracy ultrasonic flow measurement instrumentation.

PROPOSED CHANGE

The specific document changes required to support the requested increase in the licensed RATED THERMAL POWER level include both the Operating License and three affected technical specifications. The changes are:

> Revise Sections C.1 and C.2 on page 4 of the Waterford 3 Operating License, NPF-38. Section C.1 refers to the authorized maximum power level; currently at 3,390 megawatts thermal (MWt), it is proposed to be changed to 3,441 MWt. Section C.2 includes a reference to the current License Amendment number that should be updated to reflect the new amendment reflecting the approval of this change.
>
> Technical Specification 1.24 in the Definitions section of the Technical Specifications also explicitly refers to the value of RATED THERMAL POWER. The value 3,390 should be revised here to also reflect 3,441.

Entergy has conducted a review to identify if other Operating License or Technical Specification changes are needed. The conclusion of that review is that there are no additional changes to accommodate the change in the definition of RATED THERMAL POWER. Markups of the affected pages described above are provided in Attachment 3.

BACKGROUND

On June 1, 2000, a revision to 10CFR50, Appendix K was issued to be effective on July 31, 2000. The stated objective of this rulemaking was to reduce an unnecessarily burdensome regulatory requirement. Appendix K was originally issued to ensure an adequate performance margin of the Emergency Core Cooling System (ECCS) in the event a design-basis Loss of Coolant Accident (LOCA) was to occur. The margin is provided by conservative features and requirements of the evaluation models and by the ECCS performance criteria. The original regulation did not require the power measurement uncertainty be demonstrated, but rather mandated a 2% margin. The new rule allows licensees to justify a smaller margin for power measurement uncertainty. Because there will continue to be substantial conservatism in other Appendix K requirements, sufficient margin to ECCS performance in the event of a LOCA will be preserved.

However, the final rule, by itself, did not allow increases in licensed power levels. Because the licensed power level for a plant is a technical specification limit, proposals to raise the licensed power level must be reviewed and approved under the license amendment process. The license amendment request should include a justification of the reduced power measurement uncertainty and the basis for the modified ECCS analysis. These items are addressed in Attachment 2.

The resultant power increases are relatively small increases on the order of 1% to 1.5%, depending on the demonstrated instrument accuracy. Waterford 3 will be using a highly accurate ultrasonic flow measurement instrument manufactured by Caldon, Inc. The device to be used is the LEFM CheckPlus system that has been demonstrated to support a power increase of up to 1.5%. A Topical Report, ER-157P, providing a detailed description of the system and a justification of its measurement accuracy was provided for NRC review on July 6, 2001 (Letter number CNRO-2001-00029). Additional details regarding the CheckPlus system and its application at Waterford 3 are provided in Attachment 2.

BASIS FOR PROPOSED CHANGE

The basis for the proposed change is provided in Attachment 2, which documents the results of reviews of the systems, analyses, and related design topics potentially affected by the increase in operating power level.

PRECEDENTS

Similar amendment requests have been approved for:

Facility	Amendment #(s)	Approval Date	Accession #
San Onofre 2 & 3	180, 171	July 6, 2001	ML011870421
Watts Bar	31	January 19, 2001	ML010260074

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. (Entergy) is proposing that the Waterford 3 Operating License be amended to reflect an increase in the licensed reactor power level from 3,390 MWt to 3,441 MWt. These changes result from increased feedwater flow measurement accuracy to be achieved by utilizing high accuracy ultrasonic flow measurement instrumentation. The basis for this change is consistent with the revision to 10CFR50 Appendix K issued in June 2000.

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. **Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The comprehensive analytical efforts performed to support the proposed change included a review of the Nuclear Steam Supply System (NSSS) systems and components that could be affected by this change. All systems and components will function as designed, and the applicable performance requirements have been evaluated and found to be acceptable.

   The primary loop components (reactor vessel, reactor internals, control element drive mechanisms, loop piping and supports, reactor coolant pumps, steam generators, and pressurizer) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. The Leak Before Break analysis conclusions remain valid, and thus the limiting break sizes determined in this analysis remain bounding. All of the NSSS will still perform the intended design functions during normal and accident conditions. The auxiliary systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS and Balance of Plant (BOP) interface systems will continue to perform their intended design functions. The main steam safety valves (MSSVs) will provide adequate relief capacity to maintain the steam generator pressures within design limits. The atmospheric dump valves and steam bypass valves meet design sizing requirements at the uprated power level. The current Loss of Coolant Accident (LOCA) hydraulic forcing functions are still bounding for the proposed 1.5 percent increase in power.

Because the integrity of the plant will not be affected by operation at the uprated condition, it is concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions. The reduced uncertainty in the flow input to the power calorimetric measurement allows the current safety analyses to be used, without change, to support operation at a core power of 3,441 megawatts thermal (MWt). As such, all Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. Those analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to either bound operation at the 1.5 percent uprated condition, or new analyses were performed to verify all acceptance criteria continue to be met.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. **Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed changes. The new installation of the LEFM CheckPlus system has been analyzed, and failures of this system will have no effect on any safety-related system or any systems, structures or components required for transient mitigation. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?**

Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the required acceptance criteria.

Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based on the reasoning presented above, Entergy has determined that the requested change does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT EVALUATION

An evaluation of the proposed amendment has been performed pursuant to 10CFR51.22(b), and has determined that the criteria for categorical exclusion set forth in 10CFR 51.22(c)(9) of the regulations are met. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. This change does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure.

ATTACHMENT 2

TO

W3F1-2001-0091

BASIS FOR PROPOSED CHANGE

IN THE MATTER OF AMENDING

LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

Entergy Operations, Inc.

Waterford 3 Steam Electric Station

1.5 Percent Power Uprate Program

Basis for the Proposed Technical Specification Change Request

September 2001

TABLE OF CONTENTS

SECTIONS

TABLES

LIST OF ACRONYMS

AC	Alternating Current
AFW	Auxiliary Feedwater
ANSI	American National Standards Institute
ART	Adjusted Reference Temperature
ADV	Atmospheric Steam Dump Valve
AOO	Anticipated Operational Occurrences
AOR	Analysis of Record
AOT	Allowed Outage Time
ASGT	Asymmetric Steam Generator Transient
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AVB	Anti-Vibration Bar
BHP	Brake Horsepower
B&PV	Boiler and Pressure Vessel
BOC	Beginning Of Cycle
BOP	Balance Of Plant
BOL	Beginning Of Life
CCS	Containment Cooling System
CE	Combustion Engineering
CEA	Control Element Assembly
CEAW	Control Element Assembly Withdrawal
CEDM	Control Element Drive Mechanism
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CLH	Capped Latch Housing
COLSS	Core Operating Limit Supervisory System
CPC	Core Protection Calculator
CPCS	Core Protection Calculator System
CRDM	Control Rod Drive Mechanism
CVC	Chemical and Volume Control
CVCS	Chemical and Volume Control System
CW	Circulating Water
CY	Cycle
DC	Direct Current
DNB	Departure from Nucleate Boiling
DWST	Demineralized Water Storage Tank
E/C	Erosion/Corrosion
ECCS	Emergency Core Cooling System
EFPY	Effective Full-Power Year
EOC	End Of Cycle
EOI	Entergy Operations, Inc.
EOL	End Of Life

EPRI	Electric Power Research Institute
EQ	Equipment Qualification
ERG	Emergency Response Guideline
ESDR	Engineered Safeguards Design Rated
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
FER	Final Environmental Report
FR	Federal Register
FW	Feedwater System
FWCS	Feedwater Control System
GDC	General Design Criteria
gpm	Gallons Per Minute
GL	Generic Letter
HFP	Hot Full Power
HD	Heater Drain System
HPPT	High Pressurizer Pressure Trip
HPSI	High Pressure Safety Injection
HZP	Hot Zero Power
ID	Inner Diameter
IFM	Intermediate Flow Mixer
IOSGADV	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve
LAR	License Amendment Report
LBB	Leak Before Break
LBLOCA	Large-Break Loss-Of-Coolant Accident
LCO	Limiting Condition for Operation
LEFM	Leading Edge Flow Meter
LEFM✓+	Leading Edge Flow Meter CheckPlus
LHR	Linear Heat Rate
LOCA	Loss-Of-Coolant Accident
LOOP	Loss of Offsite Power
LPD	Local Power Density
LPDES	Louisiana Pollutant Discharge Elimination System
LPL	Licensed Power Limit
LPSI	Low Pressure Safety Injection
LTC	Long Term Cooling
LTOP	Low Temperature Overpressure Protection
M & E	Mass and Energy
M_{steam}	Mass Flowrate (Steam)
MFRV	Main Feedwater Regulating Valve
MHA	Maximum Hypothetical Accident

MSLB	Main Steam Line Break
MPT	Main Power Transformer
MS	Main Steam System
MSIS	Main Steam Isolation Signal
MSIV	Main Steam Isolation Valve
MSSV	Main Steam Safety Valve
Mva	Megavolt-ampere
MWe	Megawatt Electric
MWt	Megawatt Thermal
NOP	Normal Operating Pressure
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OD	Outer Diameter
ODSCC	Outer Diameter Stress Corrosion Cracking
PAC	Process Analog Control
P_{steam}	Steam Pressure
P & I	Proportional and Integral
PMC	Plant Monitoring Computer
POL	Power Operating Limit
PPS	Plant Protection System
PSV	Primary Safety Valve
P/T	Pressure/Temperature
PTE	Periodic Test Errors
PTS	Pressurized Thermal Shock
PVNGS	Palo Verde Nuclear Generating Station
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPCS	Reactor Power Cutback System
RPS	Reactor Protective System
RRS	Reactor Regulating System
RTD	Resistance Temperature Detector
RTP	Rated Thermal Power
RT_{PTS}	Pressurized Thermal Shock
RPS	Reactor Protective System
RWSP	Refueling Water Storage Pool
SAFDL	Specified Acceptable Fuel Design Limit
SBCS	Steam Bypass Control System
SBLOCA	Small-Break Loss-Of-Coolant Accident
SBO	Station Blackout

SDC	Shutdown Cooling
SDCS	Shutdown Cooling System
SFPCC	Spent Fuel Pool Cooling and Cleanup
SG	Steam Generator
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SIS	Safety Injection System
SIT	Safety Injection Tank
SLB	Steam Line Break
SSE	Safe Shutdown Earthquake
SSST	System Station Service Transformer
T_{avg}	Vessel Average Temperature
T_{cold}	Vessel/Core/Inlet Temperature
T_{hot}	Vessel Outlet Temperature
T_{steam}	Steam Temperature
TDF	Thermal Design Flow
TLU	Total Loop Uncertainty
TRM	Technical Requirements Manual
TS	Technical Specification
UFM	Ultrasonic Flow Measurement
UFSAR	Updated Final Safety Analysis Report
USE	Upper Shelf Energy
USSI	Unit Station Service Transformer
V&V	Verification and Validation
VCT	Volume Control Tank
VOPT	Variable Over Power Trip
WSES	Waterford Steam Electric Station Unit No. 3

1.0 BACKGROUND AND REASON FOR THE PROPOSED CHANGE

Waterford 3 is presently licensed for a Rated Thermal Power of 3,390 MWt. Through the use of more accurate feedwater flow measurement equipment, approval is sought to increase this core power by 1.5 percent to 3,441 MWt. The impact of a 1.5 percent core power uprate for applicable systems, components, and safety analyses has been evaluated.

This Entergy Operations, Inc. 1.5 percent core power uprate for Waterford 3 is based on eliminating unnecessary analytical margin originally required of emergency core cooling system (ECCS) evaluation models performed in accordance with the requirements set forth in the Code of Federal Regulations (CFR) 10CFR50, Appendix K (Emergency Core Cooling System Evaluation Models, ECCS).

The Nuclear Regulatory Commission (NRC) recently approved a change to the requirements of 10CFR50, Appendix K (as revised by the Federal Register (FR) 65 FR 34913, June 1, 2000). The change provides licensees with the option of maintaining the 2-percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation. For the reduced margin for ECCS evaluation case, the proposed alternative reduced margin must account for uncertainties due to power level instrumentation error. Based on the proposed use of the Caldon Leading Edge Flow Meter CheckPlus (LEFM CheckPlus) instrumentation with a power measurement uncertainty of less than 0.5 percent, it is proposed to reduce the licensed power uncertainty required by 10CFR50, Appendix K. This results in the proposed increase of 1.5 percent in the Waterford 3 licensed power level using current NRC approved methodologies.

The basis for the amendment request is that the Caldon instrumentation provides a more accurate indication of feedwater flow (and correspondingly reactor thermal power) than assumed during the development of Appendix K requirements. Complete technical support for this conclusion is discussed in detail in Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," (Reference 1.0-1) as approved in NRC's Safety Evaluation for TU Electric, dated March 8, 1999, and supplemented by Caldon Engineering Report ER-157P, Revision 3, (Reference 1.0.2). The improved thermal power measurement accuracy eliminates the need for the full 2 percent power margin assumed in Appendix K, thereby increasing the thermal power available for electrical generation.

The desired power increase of 1.5 percent will be accomplished by increasing the electrical demand on the turbine generator. As a result of this demand increase, steam flow will increase and the resultant steam pressure will decrease. The RCS nominal cold leg temperature will remain constant and the hot leg temperature will increase in response to the increased steam flow demand. The reactor coolant system (RCS) average temperature will increase slightly.

New procedures for maintenance and calibration of the LEFM CheckPlus system will be developed per the design control process based on the vendor's recommendations. Should the LEFM CheckPlus system be unavailable, the main steam or feedwater flow venturis will be used to sense flow rate in the Core Operating Limit Supervisory System, as was done prior to the installation of the LEFM CheckPlus. If the LEFM CheckPlus system is not operable the Power Limit will be administratively controlled at a level consistent with the accuracy of the available instrumentation as described in Section 3.2 below. The power limit reduction requirements, for the LEFM CheckPlus out of service, will be incorporated into the Waterford 3 Technical Requirements Manual.

2.0 DESCRIPTION OF THE PROPOSED CHANGE

The proposed license amendment would revise the Waterford 3 Operating License and Technical Specifications (TSs) to reflect an increase in core power level by 1.5 percent to 3,441 MWt. The power uprate is based on the use of the Caldon Leading Edge Flow Meter CheckPlus for determination of main feedwater flow and the associated determination of reactor power through the performance of the power calorimetric currently required by Waterford 3 TSs. Specifically, the proposed changes are provided by the markups of the current Waterford 3 operating license and TSs, in Attachment 1 of Waterford 3 Technical Specification Change Request NPF-38-238.

Entergy Operations, Inc. notes that various Combustion Engineering topical reports that are a part of the Waterford 3 licensing basis (e.g., CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model", CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model", etc.), consistent with 10CFR50 Appendix K (Reference 2.0-1) may have included explicit references to their use of "102% of licensed core power levels". These topical reports describe the Nuclear Regulatory Commission (NRC) approved methodologies which support the Waterford 3 safety analyses, including the small break and large break loss of coolant accident analyses. Along with the proposal to increase the reactor thermal power to 3,441 MWt, Entergy Operations, Inc. requests continued use of these topical reports. Entergy does not consider that these topical reports require revision to reflect this requested power uprate. Rather, it will be understood that those statements refer to the Appendix K margin and the original licensed power level. Entergy Operations, Inc. proposes that these topical reports be approved for use consistent with this license amendment request, and further, the NRC acknowledge that the change in the power uncertainty does not constitute a significant change, as defined in 10CFR50.46 and 10CFR50 Appendix K, to these topical reports.

3.0 SAFETY ANALYSIS

3.1 APPROACH

The Appendix K Power Uprate Program for Waterford 3 as described herein addresses nuclear steam supply system (NSSS) performance parameters, design transients, systems, components, accidents, and nuclear fuel as well as interfaces between the NSSS and balance-of-plant (BOP) systems. No new analytical techniques have been used to support the Appendix K power uprate project. The key points include the use of:

- Well-defined analysis input assumptions/parameter values
- Currently approved analytical techniques
- Applicable licensing criteria and standards

The evaluations and analyses described herein have been completed consistent with an increase in licensed core power from 3,390 MWt to 3,441 MWt. Section 3.3 of this report discusses the revised NSSS design thermal and hydraulic parameters that were modified as a result of the 1.5 percent uprate and that serve as the basis for all of the NSSS analyses and evaluations. Section 3.4 concludes that no design transient modifications are required to accommodate the revised NSSS design conditions. Sections 3.5 through 3.7 present the systems (e.g., Safety Injection (SI), Shutdown Cooling (SDC), and control systems) and components (e.g., reactor vessel, pressurizer, reactor coolant pumps (RCPs), steam generator, and NSSS auxiliary equipment) evaluations completed for the revised design conditions. Section 3.8 summarizes the effects of the uprate on the BOP (secondary) systems based upon a heat balance evaluation. Section 3.9 provides an analysis of the effects of the power uprate on the Waterford 3 electrical power systems. Section 3.10 provides the results of the accident analyses and evaluations performed for the loss-of-coolant-accident (LOCA) and non-LOCA transients. Sections 3.11 and 3.12 summarize the containment accident analyses and evaluations and the radiological consequence evaluations. Section 3.13 contains the results of the fuel-related analyses. The results of all of the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met.

3.1.1 General Licensing Approach for Plant Analysis Using Plant Power Level

The reactor core and/or NSSS thermal power are used as inputs to most plant safety, component, and system analyses. These analyses generally model the core and/or NSSS thermal power in one of four ways.

First, some analyses apply an explicit 2 percent increase to the initial condition power level to account solely for the power measurement uncertainty. These analyses have not been reperformed for the requested 1.5 percent uprate conditions because the sum of increased core power level (1.5 percent) and the decreased power measurement uncertainty (less than 0.5 percent) falls within the previously analyzed conditions.

The power calorimetric uncertainty calculation described in Section 3.5.10 indicates that with the LEFM CheckPlus devices installed, the power measurement uncertainty (based on a 95 percent probability at a 95 percent confidence interval) is less than 0.5 percent. Therefore, these analyses only need to reflect a 0.5 percent power measurement uncertainty. Accordingly, the existing 2 percent uncertainty can be allocated such that 1.5 percent is applied to provide sufficient margin to address the uprate to 3,441 MWt, and 0.5 percent is retained in the analysis to still account for the power measurement uncertainty.

Second, some analyses employ a nominal initial condition power level. These analyses have either been evaluated or re-performed for the 1.5 percent increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.5 percent conditions.

Third, some of the analyses already employ an initial condition power level in excess of the proposed 3,441 MWt. These analyses were previously performed at a higher power level as part of prior plant programs. For these analyses, some of this available margin has been used to offset the 1.5 percent uprate. Consequently, the analyses have been evaluated to confirm that sufficient analysis margin exists to envelope the 1.5 percent uprate.

Fourth, some of the analyses are performed at zero-percent initial condition power conditions or do not actually model the core power level. Consequently, these analyses have not been reperformed since they are unaffected by the core power-level.

3.2 FEEDWATER FLOW AND ENERGY MEASUREMENT UNCERTAINTY REDUCTION

The power uprate is based on the reduction of feedwater flow and energy measurement uncertainty. Reduction of main feedwater flow and energy uncertainty reduces the associated secondary calorimetric measurement uncertainty which is used to determine reactor power. Feedwater flow measurement uncertainty is reduced by using Caldon LEFM CheckPlus flow meters.

The Caldon LEFM CheckPlus units used at Waterford 3 are chordal transit time meters. These units measure the time required for an ultrasonic pulse to travel across a pipe from one transducer to another along a chordal path that is diagonal to the fluid flow. The difference in times of flight for pulses traveling with and against the fluid flow is proportional to the fluid velocity. Volumetric fluid flow is calculated from this measured fluid velocity and known measured physical dimensions of the meter.

Each Caldon LEFM CheckPlus meter is a pre-fabricated piping spool piece consisting of two intersecting planes of transducer pairs. Each plane has four pairs of transducers. This configuration of sensor pairs in each LEFM CheckPlus meter results in precision volumetric flow measurement, which is further documented in Caldon Topical Reports ER-80P and ER-157P.

In addition to volumetric flow measurement, these meters calculate bulk feedwater temperature with much greater precision than measured by current station temperature instrumentation. Bulk temperature is determined based on a correlation between measured feedwater pressure, temperature and sound velocity.

Feedwater mass flow is calculated by multiplying the volumetric flow measurement by feedwater density. Feedwater density is determined by using the improved feedwater temperature measurement and measured feedwater pressure. The density uncertainty is reduced as a result of the reduced feedwater temperature measurement uncertainty. Consequently, feedwater mass flow uncertainty is reduced as a result of the reduction of volumetric flow uncertainty and the reduction of density uncertainty.

The reduced feedwater bulk temperature measurement uncertainty also reduces the feedwater enthalpy uncertainty. The reduced feedwater mass flow uncertainty and reduced feedwater enthalpy uncertainty result in a significantly reduced feedwater energy rate uncertainty. The reduced feedwater energy rate uncertainty results in a reduced secondary power calorimetric uncertainty, which reduces the uncertainty of reactor power.

Reactor power is calculated in the Core Operating Limit Supervisory System (COLSS), which resides in the plant monitoring computer (PMC). The inputs to the COLSS secondary calorimetric calculation include feedwater flow, feedwater temperature, steam flow, steam generator pressure, steam header pressure, and blowdown flow. The Caldon LEFM CheckPlus meters will provide the preferred feedwater flow and temperature input to COLSS. The venturi-based feedwater or main steam flow measurement and feedwater temperature element inputs will be available to COLSS for back up in the event the Caldon LEFM CheckPlus units become inoperable.

The LEFM CheckPlus feedwater mass flow and temperature input will also be used in COLSS to adjust or "calibrate" the feedwater and main steam venturi-based flow meters calculated mass flows. The LEFM CheckPlus temperature input will be used in COLSS to adjust or "calibrate" the feedwater temperature element input. The adjustments are made continuously in COLSS by comparing the Caldon LEFM CheckPlus output to the venturi and temperature element outputs. The venturi and temperature element outputs are compensated by comparison-based multipliers to match the Caldon LEFM CheckPlus output. The comparison-based multipliers are stored in memory within the COLSS program.

In the event the Caldon LEFM CheckPlus units become inoperable, the control room operators are promptly alerted by the control room annunciator and computer alarms. COLSS will automatically use the venturi and temperature element outputs, adjusted by the comparison based multipliers retrieved from memory, to continue calculating reactor power based on the secondary calorimetric. Without the Caldon LEFM CheckPlus units in operation, the comparison based multipliers are no longer continuously updated. The uncertainties of the venturi and temperature element based inputs are expected to increase over time due to drift and ambient temperature uncertainty effects. These effects will be addressed through administrative controls.

The components of the secondary calorimetric calculation comprise of the following equation:

Reactor power = RCS energy losses – RCS energy credits + energy rate exiting the steam generators – energy rate entering the Steam generators

The RCS energy losses and credits are based on COLSS addressable constants that do not change based on measured calorimetric inputs. The energy rate entering the steam generators is the product of feedwater mass flow and feedwater enthalpy. The energy rate exiting the steam generators is the sum of the product of main steam mass flow and main steam enthalpy and the product of blowdown mass flow and blowdown enthalpy.

Table 3.2-1 summarizes the uncertainties of the measured inputs to feedwater, blowdown and main steam mass flow and enthalpy.

Table 3.2-1

Secondary Calorimetric Power Measurement Uncertainty Components
(1σ normal with mean = 0, except as noted)

Parameter	**Units**	**Venturi**	**LEFM ✔ +**
Feedwater Flow Venturi ΔP	In H_2O	7.15 (0.52%)$^{+}$	*
Feedwater Mass Flow	Klbm/hr	*	10.4(0.138%)$^{+}$
Feedwater Temperature	°F	2.5	0.3
Steam Flow Venturi ΔP	In H_2O	6.77	6.77
Blowdown Flow Rate (uniform)	gpm	43.16	43.16
Steam Quality (uniform)	NA	0.002	0.002
Secondary Pressure	Psi	10.6	10.6

* In the "Venturi" configuration, COLSS uses feedwater flow venturi ΔP and feedwater temperature inputs to calculate the feedwater mass flow. In the "LEFM ✔ +" configuration, the input to COLSS will be feedwater mass flow. The power measurement uncertainty is calculated for each configuration independently.

\+ The uncertainties for venturi ΔP for the "Venturi" configuration and the feedwater mass flow for the "LEFM ✔ +" configuration are presented in percent, as well as their appropriate units, to enable a degree of comparison. The improvement in the feedwater mass flow uncertainty results from the improved accuracy of the LEFM ✔ + equipment relative to the venturi and the reduction in the feedwater temperature uncertainty.

The Waterford 3 Caldon LEFM CheckPlus units have been extensively tested and calibrated at Alden Research Laboratories to verify the meters uncertainties were within the values assumed in the secondary calorimetric uncertainty calculations.

To further ensure this reduced power measurement uncertainty is validated and maintained, the following additional actions will be performed:

- The implementing modification package specifies the affected maintenance and operating procedures that must be in place prior to declaring these units operable and raising plant power above 3,390 Mwt.

- Although its use for calorimetric input is not nuclear safety related, the system's software has been developed and will be maintained under a verification and validation (V&V) program. The V&V program has been applied to all system software and includes a detailed code review.

3.2.1 Compliance with the NRC SER

The installation of the Caldon LEFM CheckPlus flow measurement system at Waterford Unit 3 complies with Topical Report ER-80P and ER-157P. In addition to the installation requirements, the NRC identified the following criteria that must be addressed by licensees requesting a license amendment based on the Topical Reports. Waterford 3 will comply with the four criteria described below.

3.2.1.1 Criterion 1

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM CheckPlus, including processes and contingencies for inoperable LEFM CheckPlus instrumentation and the effect on thermal power measurements and plant operation.

Response to Criterion 1

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 CheckPlus system. Plant maintenance and calibration procedures will be revised to incorporate Caldon's maintenance and calibration requirements prior to declaring the LEFM CheckPlus system OPERABLE and raising power above 3,390 MWt. The incorporation of and continued adherence to these requirements will assure that the LEFM CheckPlus system is properly maintained and calibrated.

The LEFM CheckPlus operability requirements will be contained in the Waterford 3 Technical Requirements Manuals (TRM). A Limiting Condition for Operation (LCO) has been drafted for inclusion in the TRM stating that an operable Leading Edge

Flow Meter (LEFM CheckPlus) 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 CheckPlus 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 reduced and maintained to a level that accounts for the appropriate instrumentation uncertainties thereby preserving ECCS limits.

3.2.1.2 Criterion 2

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed installation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

Response to Criterion 2

This Criterion is not applicable to the Waterford 3. Waterford 3 currently uses venturis to obtain the calorimetric heat balance measurements. Waterford 3 is installing a new LEFM CheckPlus System as the basis for the requested uprate. It will be installed during Refueling Outage Eleven.

3.2.1.3 Criterion 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.

Response to Criterion 3

The uncertainty associated with the LEFM CheckPlus and the method used to derive that uncertainty is described in Caldon topical Report ER-157P. An analysis was performed to determine and confirm the total secondary calorimetric power measurement uncertainty based on using the Caldon LEFM CheckPlus flow meters as preferred inputs and as calibration inputs to the existing feedwater and main steam venturi flow instrumentation loops. This analysis compares the uncertainties of the existing flow measurement system to the Caldon LEFM CheckPlus units.

3.2.1.4 Criterion 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.

Response to Criterion 4

Criterion 4 does not apply to Waterford 3. The calibration factor for the Waterford 3 spool pieces were established by tests of these spools at Alden Research Laboratory 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 on file. The calibration factor used for the LEFM CheckPlus at Waterford 3 will be based on these reports. The uncertainty in the calibration factor for the spools will be based on the Caldon engineering report. 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.

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 ER-80P and ER-157P). This step provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation as described in Section 3.5.10. Final commissioning is expected to be completed in April 2002.

3.3 NUCLEAR STEAM SUPPLY SYSTEM OPERATING POINT PARAMETERS

3.3.1 Introduction

The NSSS Operating Point parameters 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 flow) that are used as the basis for the NSSS analyses and evaluations. As part of the 1.5 percent increase in licensed core power from 3,390 MWt to 3,441 MWt, it was necessary to revise these parameters. Note that the operating point calculation was performed at 1.7 percent power uprate conditions (3,448 MWt) to bound the requested uprate power of 1.5 percent. The new parameters are identified in Table 3.3.1-1. These parameters have been incorporated, as required, into the applicable NSSS systems and components evaluations, as well as safety analyses, performed in support of the uprate.

3.3.2 Input Parameters and Assumptions

The NSSS Operating Point parameters are determined based on best estimate inputs, such as best estimate RCS flow, core inlet temperature and projected steam generator tube plugging (SGTP) levels, which yield primary and secondary-side conditions that best indicate the way the plant operates now and after the power uprate is in place.

The modified input assumptions include the increased NSSS power level of 3,448 MWt, increased feedwater temperature, and a slight adjustment to SG blowdown flow. Tube plugging was assumed to be a projected 500 tubes per steam generator (SG), which is also the value used in the current accident analyses. These were the only input assumptions that changed in the calculation of the NSSS operating parameters. Section 3.3.3 shows the effects of these modified input assumptions on the NSSS operating parameters.

The current cycle (CY11) operating point parameters were also calculated using selected average plant data. From this CY 11 data, the SG heat transfer coefficients were tuned to actual current plant parameters, so that the resulting power uprate operating point is accurate.

3.3.3 Results of Parameter Cases

Table 3.3.1-1 summarizes the NSSS operating point parameter case that was developed and used as the basis for the uprating project. A description of the uprated case follows.

The Appendix K Power Uprate Operating Point represents the uprated power condition with the current core inlet temperature of 545.0°F and a nominal SG level. It yields the best estimate primary-side temperatures, secondary-side steam generator steam temperature, steam pressure, and steam flow.

The bounding 1.7 percent uprate, 3,448 MWt, results in small changes to some of the NSSS design parameters. These small changes occur based on a calculation of the steam generator and secondary-side performance resulting from the increased core power. As a result of greater power coming from the steam generator, a higher steam flow is required along with a reduced enthalpy difference between the steam exiting the steam generator and the feedwater entering the steam generator. This latter effect results in a lower steam temperature and pressure.

3.3.4 Conclusions

The various NSSS analyses and evaluations described in this document use the uprated Operating Point and current design parameters appropriate for the given analytical area. The changes seen in plant parameters from the current to the uprated operating point are commensurate with the 1.7 percent power increase which bounds the requested 1.5 percent power uprate.

Table 3.3.1-1
NSSS Original Design and Appendix K Power Uprate Nominal Operating Parameters for Waterford 3

Parameter	Original Design Conditions	Appendix K Power Uprate Nominal Operating Point
Core Power MWt (input)	3,390	3,448 [2]
No. of plugged tubes per SG	50	500
Primary Bulk T_h, ^{0}F	611	600.2
Primary T_c, ^{0}F	553	545
Primary ΔT, ^{0}F	58	55.2
Primary Flow Rate, lbm/sec (input)	41,111.1	44,522.4 [3]
Primary Pressure, psia	2250	2250
Feedwater Temperature, ^{0}F	445	442.7
Feedwater Enthalpy, BTU/lbm (input)	424.9	422.2
FW Flow Rate per SG, lbm/sec	Same as Steam Flow	2,135.9
SG Blowdown Flow per SG, lbm/sec (input)	NA	17.48
SG Steam Flow per SG. lbm/sec	2,097.2	2,118.4
Steam Pressure, psia	900	831.5
SG Total Mass, lbm	176,950 [1]	174,030 [4]
SG Liquid Mass (lbm)	163,844	159,158

[1] Does not include mass in steam lines from SG to MSIV (approximately 2500 lbm)
[2] This value of Core Power used for analysis purposes only as described in Section 3.3.1
[3] Appendix K Uprate Operating Point Flow based on Actual Pump Performance
[4] Includes mass in steam lines from SG to MSIV (approximately 2500 lbm)

3.4 DESIGN TRANSIENTS

The main 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. With respect to the type of transients and frequency of occurrence, the implementation of the Appendix K power uprate will not create new types of transients nor increase the probability of occurrence of any design transients.

The existing design transients represented conservative estimates strictly for design purposes and were not intended to be accurate representations of actual transients. These conservative estimates allowed for additional margin. In the case of Upset and Emergency Conditions, the transients are initiated from 102% power. For these reasons alone, the types and frequency of transients listed in the existing design transients documents remain valid. However, the impact of the changes on existing design conditions, due to the Appendix K uprate, were evaluated against the pressure and temperature transient assumed for each of the design transients.

The detailed evaluation verified that the original design transients were conservatively developed with respect to the rate, and extent, of pressure/temperature changes during 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 (i.e., maximum flowrates), and are therefore, not impacted by the uprate. For the more severe type transients (emergency, upset and faulted conditions), the evaluations were initially based on 102% reactor power to begin with or evaluation of NSSS Control System Setpoint Transients demonstrated the original design transients were conservatively specified. The Appendix K uprate does not impact the frequency of occurrence for any of the transients. Therefore, the thermal-hydraulic transients in the original specifications still remain valid with the Appendix K uprate for Waterford 3.

3.5 NUCLEAR STEAM SUPPLY SYSTEMS

This section presents the results of the evaluations and analyses performed in the NSSS area to support the revised operating conditions provided previously in Section 3.3. The systems addressed in this section include fluid systems and control systems. The results and conclusions of each evaluation and analysis are presented within each subsection.

3.5.1 Reactor Coolant System

The purpose of the Reactor Coolant System (RCS) is to remove heat from the core and transfer it to the secondary side of the steam generators. The RCS consists of the reactor pressure vessel, two hot leg pipes, two steam generators, four reactor coolant pumps, four cold leg pipes and one pressurizer with attendant interfacing piping, valves and instrumentation.

Various assessments were performed to ensure that the RCS design basis functions could still be met at the revised operating conditions. The principal effects of power uprate on the RCS are a slight increase in T_{hot} and the increase in decay heat. The normal operating pressure of 2250 psia remains unchanged. The results of the evaluation of uprated conditions on the RCS functions are described below:

a. The increase in T_{hot} will increase the total amount of heat transferred to the main steam system (MSS). Verification that the major components of the nuclear steam supply system can support this increase in the normal heat removal function is addressed in Sections 3.7 and 3.8.

b. The increased thermal power can change the transient response of the RCS to normal and postulated design basis events. The acceptability of the RCS with respect to control and protection functions is addressed in Sections 3.5.6 and 3.10.3.

c. The cold leg temperature remains unchanged at a nominal value of 545 °F. As a result, the RCS mass flow is not affected by the uprate.

d. Reactor coolant system design temperature and pressure of 650 °F and 2500 psia continue to remain applicable for the uprate conditions.

e. The pressurizer design temperature and pressure of 700 °F and 2500 psia continue to remain applicable for the uprate conditions.

f. 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 (Section 3.10.3).

3.5.2 Safety Injection System

The function of the Safety Injection (SI) System is to remove the stored energy and fission product decay heat from the reactor core following a loss-of-coolant accident. The system is designed such that fuel rod damage is minimized, facilitating the long-term removal of decay heat. The system also provides injection of negative reactivity (boron) in the RCS cooldown events such as a main steam line break.

The active part of the SI System consists of high pressure safety injection (HPSI) pumps, the refueling water storage pool (RWSP), low pressure safety injection (LPSI) pumps, and the associated valves, instrumentation, and piping.

The passive portion of the SI System is the Safety Injection Tanks (SIT) that are connected to each of the RCS cold leg pipes. Each safety injection tank contains borated water under nitrogen pressure, and automatically injects into the RCS when the RCS pressure drops below the operating pressure of the SITs. The active portion of the SI System (injection pumps) injects borated water from the RWSP into the reactor following a break in either the RCS or steam system piping to cool the core and prevent an uncontrolled return to criticality.

The SI System is described in two phases; the injection phase and the recirculation phase. The injection phase provides emergency core cooling and additional negative reactivity immediately following a spectrum of accidents including LOCA by prompt delivery of borated water to the reactor vessel. The recirculation phase provides long-term post-accident cooling by recirculating water from the containment sump.

During normal operation the SI System does not operate and has no design function. Thus, during normal operation, there is no impact on the system due to the proposed power uprate. However, the slight increase in RCS stored energy and decay heat resulting from the power uprate are well within the capabilities of the SI System to respond to design basis events. The results of the evaluation of a Loss of Coolant Accident are presented in Section 3.10.2. For non-LOCA RCS depressurization events, the SI System is acceptable for power uprate as demonstrated in Section 3.10.3.

3.5.3 Chemical and Volume Control System

The chemical and volume control system (CVCS) provides 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, 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 reactor coolant system in the cold leg, downstream of the reactor coolant pump.

The nominal T_{cold} for the power uprate remains unchanged at 545.0 °F. As a result, the temperature of the letdown flow is not changed. Consequently, there is no impact on the thermal performance of the CVCS.

The CVCS provides a source of borated water for post accident injection. Evaluation of required Emergency Core Cooling System (ECCS) water volumes and boric acid concentrations will be performed as part of the normal Reload Safety

Evaluation process. The slight increase of N-16 activity at uprate conditions has a negligible effect on letdown line delay time requirements. There will be no change to the letdown and makeup requirements as a result of power uprate.

As previously noted, T_{cold} and the reactor coolant mass flow rate remain unchanged. Increased power is due to a slight increase in T_{hot} and associated increase in 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.

3.5.4 Shutdown Cooling System

The Shutdown Cooling (SDC) System is designed to remove sensible and decay heat from the core and to reduce the temperature of the RCS during the second phase of plant cooldown.

The SDC System consists of two trains. Each train consists of one heat exchanger, one LPSI pump, associated piping, valves, and instrumentation. Each train takes suction from one reactor coolant hot leg, flows through the LPSI pump, the tube side of the SDC heat exchanger, and back to the two associated RCS cold legs.

The Waterford 3 SDCS was previously evaluated for an 8% power uprate. The evaluation consisted of normal two train and single train cooldown to cold shutdown conditions and refueling conditions, and a single train RSB 5-1 cooldown to 200 °F. The analysis performed for the 8% power uprate demonstrates that the Shutdown Cooling System is capable of performing the required functions for the normal operation and safe shutdown conditions.

In addition to the above, Technical Specifications Surveillance Requirements 4.9.8.1 and 4.9.8.2 require a minimum reactor coolant flow rate of 4000 gpm with an allowable reduction to 3000 gpm 175 hours after shutdown. This reduction in flow is necessary to address vortexing concerns during reduced inventory operations. The Technical Specifications allow a further reduction to 2000 gpm after 375 hours. The results of the analysis performed for the 8% power uprate showed that the required time for a reduction in flow to 3000 gpm remains bounding. The required time for a reduction to 2000 gpm remains unchanged.

3.5.5 Containment Cooling

The Containment Fan Cooling Subsystem (CCS) is designed for use during normal and post-accident operation. During normal operation the CCS provides cooling to various areas of containment.

The post-accident safety related design functions of the CCS are as follows:

1. Remove heat from the containment atmosphere following a loss of coolant accident (LOCA), secondary system pipe rupture, or main steam line break (MSLB) inside containment.

2. Maintain an acceptable containment pressure and temperature.

3. Limit off site radiation dose by reducing the pressure differential between containment atmosphere and the external environment.

The purpose of the Containment Spray (CS) System is to remove heat during and following an accident which involves either a Loss of Coolant Accident (LOCA), a secondary system pipe rupture or a Main Steam Line Break (MSLB) inside containment. The spray will also reduce the containment pressure. By reducing the differential pressure between the containment atmosphere and the external environment, the driving force for fission product leakage across the containment is reduced. This action will limit offsite radiation by the reduction of iodine in the post accident containment atmosphere.

The containment peak pressure / temperature analyses and the radiological consequence calculations were performed at or above 102 % power. Thus the Appendix K uprate will remain bounded by the existing analysis.

3.5.6 NSSS Transient Control Systems and Components

Entergy Operations, Inc. began operating Waterford 3 at reduced RCS temperatures for Cycle 6. ABB Combustion Engineering Nuclear Power (now Westinghouse Electric Company, LLC) provided a study to determine optimum NSSS control system setpoints for operation at reduced RCS temperatures. That study provided a transient evaluation to confirm that the plant can respond appropriately to the following transients without generating a reactor trip or engineered safety feature actuation system (ESFAS) actuation. These same transients were evaluated for the revised operating point of 100% power including a maximum power uprate of 1.7%. The transients of concern include:

- 10-percent step load decrease from 100-percent power
- 100-percent power loss of main feedwater pump
- 100-percent power turbine trip
- 5-percent per minute ramp load decrease from 100-percent power

The analysis methodology for these transients employs best estimate analysis using the projected operating point for a maximum of 1.7 percent power increase. Both beginning-of-cycle (BOC) and end-of-cycle (EOC) fuel reactivity conditions were considered.

The results of the transient analysis study provides recommended NSSS control system setpoints for the Steam Bypass Control System (SBCS), the Reactor Power Cutback System (RPCS), the Feedwater Control System (FWCS) and the Reactor Regulating System (RRS). The transients of concern above were evaluated using both the current NSSS control system setpoints and the recommended NSSS control system setpoints. In conclusion, the current NSSS control system configuration, with either the current setpoints or the setpoints modified slightly for the changes in the operating point, will respond to the above transients from the new 100% power operating point without generating a reactor trip or engineered safety feature actuation system actuation.

3.5.7 Low Temperature Overpressurization Protection Relief Valves

Low Temperature Overpressure Protection (LTOP) for the WSES-3 plant is provided by the two relief valves located in the Shutdown Cooling System suction lines in conjunction with specific operational controls, e.g., heatup and cooldown rates. Together, these measures are designed to protect the RCS from brittle fracture from overpressure events when one or more of the RCS cold legs are at temperatures less than or equal to 272°F. The increase in core power due to the uprate will have a corresponding effect on decay heat, which is expressed in terms of a fraction of rated thermal power.

Two transients are analyzed in support of LTOP: mass addition and energy addition. The limiting LTOP transient is the mass addition transient (inadvertent Safety Injection Actuation Signal, SIAS). That transient includes the mass addition from three charging pumps and two HPSI pumps as well as the heat input from all pressurizer heaters. Decay heat was not used as an input in the calculation of record. The original sizing calculation for the Shutdown Cooling System suction relief valves included the above parameters plus the third HPSI pump, providing an additional 20% margin. The peak pressure for the mass addition transient is the opening pressure (set pressure plus accumulation), which demonstrates the excess capacity of the valves under the limiting transient.

Decay heat is an input to the energy addition transient at Waterford 3 (Reactor Coolant Pump, RCP, start with hot steam generators). In the energy addition transient, heat input to the RCS is from the steam generator secondary side with additional input from the pressurizer heaters, decay heat (1% of rated thermal power) and RCP joule heat. An increase of 1.5 percent in the rated thermal power, to 3,441 MWt, will slightly increase the heat input in this transient. The increased heat will cause the valve to reach its opening pressure slightly earlier in the transient, however, due to the excess capacity, there will be no increase in the peak pressure for this transient.

The 10 CFR 50, Appendix G P/T curves and reference temperature values do not change as a result of the power uprate (refer to Section 3.6.4). Since the existing peak pressure for the limiting LTOP analysis is unaffected and the P-T curves and existing setpoints are not affected, continued Low Temperature Overpressure Protection is ensured.

3.5.8 Plant Protection System

The Waterford 3 Plant Protection System (PPS) is comprised of an Engineered Safety Features System (ESFAS) and a Reactor Protection System (RPS).

The ESFAS consists of sensors, logic and other equipment necessary to monitor selected NSSS and containment conditions in order to generate signals to actuate the ESF and ESF support systems. The ESFAS uses inputs from the Process Analog Control (PAC) System cabinet, processes the signals and generates outputs to the Auxiliary Relay Cabinet. The ESFAS logic circuits are located in the PPS cabinet.

The Reactor Protective System (RPS) is that portion of the PPS which generates signals that actuate reactor trip. The RPS consists of sensors, calculators, logic and other equipment necessary to monitor selected Nuclear Steam Supply System (NSSS) and containment conditions and to effect reliable and rapid CEA insertion (reactor trip) if any or a combination of the monitored conditions approach specified safety system settings.

The uprate does not impact or modify any of the PPS hardware. The potential impact of the uprate on PPS setpoints is discussed in detail in section 3.10.1.

3.5.9 Core Protection Calculators

The Core Protection Calculator System (CPCS) initiates the Low Departure from Nucleate Boiling 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 transient (ASGT) 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. No other changes to CPCS algorithms or constants are required due to the increase in rated thermal power. However, it may be necessary to adjust the VOPT setpoints based on the results of the transient analysis being evaluated for the increased power level. The most important VOPT setpoints are addressable constants so that they will be changed, as required, as part of the normal reload process when the increased RTP is implemented.

3.5.10 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). COLSS continually monitors the Technical Specifications limiting conditions for operations (LCOs) on the following:

1. Linear Heat Rate (LHR)
2. Margin to Departure from Nuclear Boiling Ratio (DNBR)
3. Total Core Power
4. Azimuthal Tilt
5. 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 increase. These constants will be calculated as part of the reload fuel design process.

COLSS measures the core thermal power based on three methods – turbine first stage pressure, primary calorimetric and secondary calorimetric – and uses it to determine margins to the power operating limits (POLs) on DNBR and LHR as well as the margin to the licensed power limit (LPL). The COLSS secondary calorimetric power is the standard by which the other power values are calibrated. The use of more accurate ultrasonic feedwater flow measurement equipment reduces the COLSS secondary calorimetric power measurement uncertainty at high power from 2 percent to less than 0.5 percent RTP. Therefore, the LPL will remain at 100 percent RTP while the RTP is increased by 1.5 percent by revising the constants that are affected by the increased RTP, including the core average heat flux and core average linear heat rate.

The COLSS secondary calorimetric power measurement uncertainty is discussed in further detail in Section 3.2.

If the ultrasonic feedwater flow measurement equipment is out of service for more than the allowed outage time (AOT), it will be necessary to reduce the LPL in COLSS (see Section 3.2).

3.5.11 Spent Fuel Pool Cooling

The spent fuel pool cooling and cleanup (SFPCC) system is designed to remove the decay heat from spent fuel assemblies stored in the spent fuel pool, and to clarify and purify the water in the spent fuel pool. The spent fuel pool cooling portion of the SFPCC system is a seismic Category I closed loop system consisting of two half capacity pumps and one full capacity heat exchanger. Heat is removed from the spent fuel pool heat exchanger by the component cooling water system. A backup fuel pool heat exchanger with a lower heat removal capacity is available for use when the spent fuel pool primary heat exchanger is out of service or in an emergency.

The spent fuel pool cooling portion of the SFPCC system was reanalyzed for the 1998 Waterford 3 Spent Fuel Pool Rerack Project (Reference 3.5.11-1). Several conservative assumptions were used for the 1998 reanalysis (core power of 3,661 MWt, two year fuel cycle, 5% enriched fuel etc.), which result in the reanalysis conservatively bounding the proposed 1.5 percent uprate. Since the 1998 reanalysis bounds the proposed uprate no changes are required to the spent fuel pool cooling portion of the SFPCC system.

3.6 NUCLEAR STEAM SUPPLY SYSTEM COMPONENTS

Reactor Coolant System (RCS) Loss of Coolant Accident (LOCA) loads are discussed in general terms and with respect to the Appendix K uprate, followed by individual discussions of the structural integrity assessments performed for the RCS major components. This section concludes with a discussion of the effects of small variations in the Appendix K Uprate Operating Point data.

3.6.1 Reactor Coolant System LOCA Forces Evaluation

The purpose of a LOCA hydraulic forces analysis is to generate the hydraulic forcing functions and blowdown loads that occur on RCS components as a result of a postulated LOCA. These forcing functions and loads act on the component's shell and internal structures.

The full set of RCS loadings considered in the structural analysis of a LOCA event consists of the internal forcing functions generated from the hydraulic forces analysis, the pipe tension release and jet impingement forces acting at the break locations, and, where applicable, the external loads due to subcompartment pressurization effects that act on the components and their supports.

In support of the 1.5 percent uprating conditions for Waterford 3, an evaluation was performed to assess the impact of the uprated RCS conditions defined in Section 3.3 on the LOCA-induced hydraulic blowdown loads. This evaluation demonstrated that, at the 1.5 percent uprate conditions of Table 3.3.1-1, the original design basis LOCA hydraulic loads in the UFSAR resulting from the mechanistic failure of main coolant loop piping would bound analogous loadings resulting from tributary line breaks typical of those associated with leak-before-break (LBB) considerations.

The LBB analyses of Reference 3.6-1 justified the elimination of large primary-loop pipe ruptures as the LOCA design basis for evaluations of structural integrity. As referenced in Reference 3.6-2, the NRC in 1990 had approved the application of LBB for structural analyses at Waterford 3, consistent with current NRC guidance.

Nevertheless, since the LOCA hydraulic loads produced by tributary line breaks would be bounded by those of the original design basis LOCA resulting from the mechanistic failure of main coolant loop piping, the following structural evaluation discussions are based on the original design basis event, and do not make direct use of the mitigating effects of LBB. Consequently, the design basis event for blowdown loads at the uprated conditions remains the original design basis LOCA.

3.6.2 RCS Major Component Assessments

3.6.2.1 Reactor Vessel Structural Evaluation

This evaluation assesses the effects that the 1.5 percent uprating conditions have on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors in each of the vessel regions, as identified in the reactor vessel stress reports and addenda.

The NSSS design transients are demonstrated to be unaffected by the 1.5 percent power uprate (see Section 3.4). Furthermore, the nominal vessel outlet temperature increases to 600.2 °F, and the nominal vessel inlet temperature remains at the current Cycle 11 value of 545.0 °F as a result of the 1.5 percent uprate program. Therefore, the T_{hot} variation during normal plant loading and plant unloading increases while the T_{cold} variation remains unchanged.

As noted above, the nominal vessel inlet temperature associated with the 1.5 percent power uprate is the same as the nominal temperature for the current cycle. The nominal vessel outlet temperature has increased slightly but is still less than the normal operating vessel outlet temperature of 611°F that was originally used in the analysis of the reactor vessel outlet nozzles. Therefore, the effects of the plant loading and unloading transients on the inlet and outlet nozzles remain bounded by the stress Analyses of Record.

The reactor vessel main closure flange region and control rod drive mechanism (CEDM) housings were originally evaluated for the effects of a higher vessel outlet temperature. Therefore, the effects of the 1.5 percent uprate vessel outlet temperature on these regions are also bounded by the current design basis.

The remaining reactor vessel regions, including the inlet nozzles, vessel wall transition, core support guides, bottom head-to-shell juncture, and instrumentation nozzles are affected by the vessel inlet temperature. However, the inlet temperature is unchanged for the 1.5 percent power uprate. Therefore, the previously determined maximum stress intensity ranges and maximum cumulative fatigue usage factors for these regions are valid.

The Code version used in the evaluation for the Waterford 3 reactor vessel, steam generators (primary side) and pressurizer is the 1971 Edition of Section III of the

ASME Boiler & Pressure Vessel Code through the Summer 1971 Addenda. The Code is the same as the current Code of Record for the respective components.

Conclusion

The reactor vessel evaluation for the 1.5 percent power uprate demonstrates that the maximum ranges of stress intensity remain within their applicable acceptance criteria, and the maximum cumulative fatigue usage factors remain below the acceptance criterion of 1.0.

In addition, the faulted condition stress analyses for the Waterford 3 reactor vessel do not change as a result of the 1.5 percent power uprate, because no changes in the faulted condition reactor vessel/reactor internals interface loads or other faulted condition parameters are identified as a result of the uprating.

3.6.2.2 Reactor Vessel Internals Evaluation

The reactor internals support the fuel and control rod assemblies, experience control rod assembly dynamic loads, and transmit these and other loads (e.g., deadweight, seismic vibration) to the reactor vessel. The internals also direct flow through the fuel assemblies, provide adequate cooling to various internals structures, and support in-core instrumentation. The changes in the RCS operating parameters identified previously in Section 3.3 produce insignificant changes in the boundary conditions experienced by the reactor internals components. This section describes the evaluation performed to demonstrate that the reactor internals can perform their intended design functions at the 1.5 percent uprated conditions.

3.6.2.2.1 Thermal Hydraulic Systems Evaluations

A key area in evaluation of core performance is the determination of the hydraulic behavior of coolant flow and its effect within the reactor internals system. The core bypass flows are required to ensure reactor performance and adequate reactor vessel head cooling. The hydraulic lift forces are critical in the assessment of the structural integrity of the reactor internals. The results of the thermal-hydraulic evaluations are provided below.

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 2.60% 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. Therefore, a lower bound core bypass flow equal to 1.5% of the reactor vessel flow was conservatively used in the hydraulic loads calculation. The best-estimate core bypass flow is 2.28% of the reactor vessel flow.

Hydraulic Loads

An assessment that bounds 1.5 percent power uprate conditions was performed. Hydraulic loads on the reactor vessel components determined in the Waterford 3 Analyses of Record were evaluated. In Addition, a later analysis performed for another CE reactor with the same reactor internal component configuration and characteristics, and adjusted core pressure losses associated with 16 x 16 fuel assemblies in CE 3,400 MWt reactors, was considered. The results of these assessments demonstrate that the existing design loads are bounding for the 1.5 percent uprate.

Control Element Assembly Drop Time Analyses

Waterford 3 Technical Specification 3.1.3.4 requires the following:

- The arithmetic average of the control element assembly (CEA) drop times of all full-length CEAs from a fully withdrawn position shall be less than or equal to 3.0 seconds.
- The individual full length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 3.2 seconds from when the electrical power to the CEA mechanism is interrupted until when the CEA reaches the 90% insertion position, with a) T_{avg} greater than or equal to 520 °F and b) all reactor coolant pumps operating.

CEA drop times are explicitly confirmed to meet the times assumed in the accident analyses. An evaluation was performed for all Combustion Engineering (CE) designed plants to demonstrate continued compliance with the current technical specification requirements based on CE's robust five finger design, which has not shown any failure to insert at any time in life through the end of life core burnup. Uprate to 3,441 MWt will slightly increase the power level in leading rodded fuel assemblies, but will not change the burnup levels of those fuel assemblies, since the excess reactivity will be depleted faster.

Furthermore, since the existing projection of fluence used in the reactor vessel design remains bounding for uprate conditions (see Section 3.6.4), fluence induced changes in grid cage structures will not be affected by the uprate. Finally, the fluid density has not increased for Appendix K Uprate since T_{cold} has not changed and T_{hot} has increased slightly. Therefore, drop times are not adversely affected by the uprate.

Based on the above, the current limiting rod drop time requirements remain valid for the 1.5 percent uprated conditions.

3.6.2.2.2 Mechanical Evaluations

As discussed previously, the 1.5 percent uprate conditions do not affect the current design bases for seismic and LOCA loads. Therefore, it was not necessary to re-evaluate the structural effects from seismic operating-basis earthquake (OBE) and safe shutdown earthquake (SSE) loads, as well as from the LOCA hydraulic and dynamic loads.

With regards to flow-and pump-induced vibration, the current analysis uses a mechanical flow that changes by less than 1 percent for the revised operating conditions. The revised operating conditions alter the T_{hot} fluid density. However, this very small change in the T_{hot} fluid density has a negligible effect on the forces induced by flow. In addition, the 1.5 percent uprate results in a negligible change in T_{avg}. Therefore, the mechanical loads are not affected by the 1.5 percent uprated conditions.

3.6.2.2.3 Structural Evaluations

Evaluations are required to demonstrate that the structural integrity of the reactor internal components is not adversely affected by the 1.5 percent uprate conditions. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be accounted for in the design and analysis of various components.

With little or no increase in thermal flow and change in the RCS nominal operating temperatures, there will be little or no change in the boundary conditions experienced by the reactor internals components. A reactor internals assessment performed for the 1.5 percent uprate conditions determined that the thermal gradients and hydraulic loads are bounded, either by previous analyses performed for a proposed 8 percent uprate at Waterford 3 or by previous analyses performed for another CE reactor with the same reactor internal component configuration and characteristics. Therefore, the structural integrity of the reactor internal components under 1.5 percent uprated conditions was demonstrated.

Increases in core thermal power will slightly increase nuclear heating rates in the reactor vessel internals, such as lower core support plate, fuel alignment plate, and core shroud. Evaluations have been performed verifying that the existing thermal-hydraulic AOR will support the Appendix K uprate. Therefore, the calculated component lifetimes will envelope the component lifetimes associated with Appendix K uprate related increases in gamma heating.

3.6.2.3 Control Element Drive Mechanisms

The CEDMs are mounted on top of the Waterford 3 reactor head. These components are affected by the reactor coolant pressure, vessel outlet temperature, and hot leg NSSS design transients.

According to Section 3.4, the current NSSS design transients remain unchanged for the 1.5 percent uprate program. In addition, the reactor coolant pressure (2,250 psia) for the 1.5 percent uprate conditions remains the same as originally specified for the CEDMs.

The best estimate operating point vessel outlet temperature for the 1.5 percent uprate will increase slightly but remains well below the design operating temperature of 611 °F. Therefore, no additional assessments of the impact of thermal loads on the CEDMS and CEDM nozzles are required. Since there are no changes in the seismic and accident load conditions, it is concluded that the CEDMs and CEDM nozzles continue to meet structural design requirements for the 1.5 percent uprate.

3.6.2.4 Pressurizer Surge Line Piping

Parameters associated with the 1.5 percent uprating were reviewed for their impact on the design basis analysis for the pressurizer surge line piping including the effects of thermal stratification. NSSS design parameters, NSSS design transients, and changes at the reactor coolant loop Auxiliary Class 1 branch nozzle connections due to deadweight, thermal, seismic, and LOCA loading conditions were considered.

As discussed in Section 3.4, the NSSS design transients are not affected by the uprating. Therefore, the design transients remain valid for the pressurizer surge line piping.

Thermal stratification takes place during plant transients (e.g., during plant heatup), and the temperature ranges defined in the stratification AOR were conservatively based on plant operating data. T_{hot} has increased slightly for the 1.5 percent uprate. This change has a negligible effect on the stratification AOR, since it only results in a slight reduction in the ΔT between the pressurizer and the hot leg during steady state normal operation. Therefore, the stratification temperature ranges developed in the AOR will bound the new operating conditions.

There is no impact on the deadweight analysis due to the 1.5 percent uprate because there is no discernable change in the weight of the Auxiliary Class 1 pressurizer surge line piping systems. Fluid weight changes due to the change in T_{hot} are very small, and their effect on the overall piping weight is insignificant. The seismic response spectra remain unchanged. Therefore, there is no impact on the seismic analysis. Section 3.6.1 determined that it is valid to continue to base RCS structural analyses on the original design basis LOCA events. Therefore no change to the LOCA hydraulic forcing functions is required. In conclusion, the Appendix K uprate has no impact on Auxiliary Class 1 branch nozzle connection loads resulting from the deadweight, thermal, seismic, or LOCA input loading conditions.

Based on the above, the existing pressurizer surge line piping analysis remains valid.

3.6.2.5 Reactor Coolant Pumps and Motors

3.6.2.5.1 Reactor Coolant Pump Structural Analysis

The four RCPs are installed in the cold legs of the reactor coolant loops. The RCPs are affected by the reactor coolant pressure, steam generator outlet temperature, and primary-side cold leg NSSS design transients. The steam generator outlet temperature affects both the thermal expansion and thermal transient loads on the RCPs.

The nominal steam generator outlet temperature identified in Section 3.3 for Appendix K Uprate is 545.0 °F, which is the same as the current nominal outlet temperature, and lower than the design basis temperature of 553 °F. Consequently, RCP thermal expansion loadings for uprate are bounded by the design condition. Furthermore, the applicable NSSS design transients and the reactor coolant pressure are unaffected by the 1.5 percent uprate. Therefore, the existing RCP stress analyses are bounding and remain applicable for the pressure boundary components.

3.6.2.5.2 Reactor Coolant Pump Motor Evaluation

A previous Waterford 3 engineering study determined that the RCP motors were acceptable for continuous operation with limiting hot loop and cold loop conditions. The RCP motors were determined to remain acceptable for operation at the 1.5 percent uprate parameters based on the following:

- No-load T_{avg} is unchanged by this uprating. Therefore, the RCP hot start is not affected.
- Limiting RCP motor starting conditions occur during RCS cold loop conditions that are not impacted by the 1.5 percent uprate.
- The loads controlling RCP motor thrust bearing design are associated with seismic and LOCA conditions (i.e., RCP motor peak accelerations). These loads are not affected by the 1.5 percent uprate.

3.6.2.6 Steam Generators

3.6.2.6.1 Steam Generator Structural Integrity

As noted in Section 3.4, the NSSS design transients are demonstrated to be unaffected by the 1.5 percent uprate. These design transients were used as input to generate the original or baseline calculations. Since the operating conditions with the 1.5 percent uprate have slightly increased, an assessment was performed to determine the effects of key loading changes on the subcomponents. In part, previously determined results for a proposed 8-percent uprate at Waterford 3 were used to support the current efforts.

The steam generator tubes were evaluated for the effects of LOCA load increases on tube degradation and for the effects of thermal hydraulic load changes on flow induced vibration. The tubesheet and related structures were assessed for the effects of increased primary to secondary ΔP. Finally, primary head divider plate, baffle and baffle support, secondary shell and feedwater nozzle fatigue was evaluated for the effects of uprate driven parameter changes.

The results of the structural evaluations demonstrate that the steam generators meet the requirements of the American Society of Mechanical Engineers (ASME) Code limits for stress and fatigue for the 1.5 percent uprate conditions.

3.6.2.6.2 Steam Generator Thermal-Hydraulic Performance

The following evaluations and analyses were performed to assess the magnitude and importance of changes in the secondary-side thermal-hydraulic performance characteristics for the Waterford 3 steam generators at the 1.5 percent power uprate conditions.

Circulation Ratio/Bundle Liquid Flow

The circulation ratio is a measure of tube bundle liquid flow in relation to the steam flow and is primarily a function of steam flow. The bundle liquid flow minimizes the accumulation of contaminants on the tubesheet and in the bundle. The 1.5 percent increase in power causes the bundle liquid flow to decrease by less than 1 percent and the circulation ratio to decrease by less than 2 percent. Therefore, the uprating and other operating condition changes have minimal effect on this function. No effect on sludge accumulation or local concentrations is expected.

Damping Factor

The hydrodynamic stability of a steam generator is characterized by the damping factor. A negative value of this parameter indicates that small perturbations in the steam pressure or circulation ratio will diminish rather than grow in amplitude, thereby promoting continued stability. An evaluation confirmed that the damping factor will have a highly negative value at uprated conditions. Therefore, the steam generators will continue to remain hydrodynamically stable.

Steam Generator Pressure Drop

The increase in total secondary-side pressure drop resulting from the uprating is approximately 1 psi. This increase is very small in relation to the total feedwater system pressure drop and will have a negligible effect on the feedwater system operation.

Moisture Carryover

The performance of the Waterford 3 moisture separator packages is primarily a function of steam flow, steam pressure, and water level. An analysis was performed to determine the effect of the power uprate on the Waterford 3 moisture carryover. This was accomplished by projecting the separator performance from field performance data for Waterford 3. From the extrapolation of the field performance data, the moisture carryover is estimated to remain no more than 0.20 percent.

3.6.2.6.3 Steam Generator Hardware Changes and Additions Evaluation

Evaluations were performed to determine the impact of the revised operating conditions for the power uprate (shown previously in Section 3.3) on the structural integrity of the steam generator hardware changes and additions. These hardware changes and additions are qualified for installation in the Waterford 3 steam generators. They consist of the mechanical and welded tube plugs. The following plugs have been used in the Waterford 3 steam generators:

1. Westinghouse Inconel 600 Preservice Tapered Welded Plug
2. Westinghouse Inconel 690 Rolled Mechanical Plug
3. Westinghouse Inconel 690 Ribbed Mechanical Plug
4. Framatome Inconel 690 Tapered Welded Plug
5. Framatome Inconel 690 Mechanical Plug

The steam generator hardware structural evaluations for the 1.5 percent uprated conditions were performed to the applicable requirements of the ASME Boiler and Pressure Vessel (B&PV) Code, Section III.

Steam Generator Tube Mechanical Plug

The Westinghouse mechanical tube plugs were evaluated for the effects of changes to the thermal transients due to the power uprate.

The Westinghouse (Ribbed / Rolled) and Framatome (Rolled) mechanical plugs were evaluated for the effects of changes due to thermal transients as a result of the power uprate. The Westinghouse and Framatome mechanical tub plugs are adequately anchored in the tubes for all steady state and transient conditions. All of the stress/allowable ratios are 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 including ASME Code Case N-474-1. Since the fatigue exemption requirements are satisfied, the usage factor will remain within the Code limit of 1.0 and an explicit calculation of the usage factor is not required.

Rolled Alloy 690 steam generator tube plugs manufactured by Westinghouse have been evaluated for the effects of an increase in the primary to secondary pressure differential resulting from the power uprate. This evaluation demonstrates the continued adequacy of the Westinghouse rolled plugs to perform their intended function while fulfilling applicable ASME Boiler and Pressure Vessel Code Section III requirements.

3.6.2.6.4 Inspection Program and Tube Repair Criteria

The applicable operating parameters for the proposed 1.5 percent uprating for Waterford 3 specify a minimum full-power steam pressure of 831.5 psia. As a result, there will be an increase in the normal full power primary to secondary pressure differential. However, this increase does not affect the tube repair criterion, and the 40% through wall tube plugging limit specified in the Technical Specifications remains valid. The increase in primary to secondary pressure differential will be accounted for and adjusted in the SG Degradation Assessment specific to in-situ pressure testing screening criteria.

The thermal-hydraulic evaluation performed also investigated the potential for increasing the number of tubes that could be affected by batwing wear. This evaluation concluded that only a small effect, if any, would be readily apparent during normally scheduled inspections. Therefore, any additional tube that may be affected would have relatively slow wear rates and would be taken out of service before tube structural limits were compromised.

With respect to the proposed 1.5 percent power uprate, the inspection program will include consideration of the higher temperatures in crack growth rate analyses. Waterford 3 presently performs 100 % inspection of those portions of the tube bundle susceptible to primary water stress corrosion cracking (PWSCC) and outer diameter stress corrosion cracking (ODSCC) due to residual stresses, deposits and influence of RCS temperature.

3.6.2.7 Pressurizer

The conditions that could affect the primary-plus-secondary stresses, and the primary plus secondary plus peak stresses, are the changes in the RCS hot leg temperature (T_{hot}), the RCS cold leg temperature (T_{cold}), and the pressurizer transients. Nominal T_{cold} is unchanged, and the increase in nominal T_{hot} is very small. A T_{hot} change of this magnitude is enveloped by the current stress analysis. Since the design transients (see Section 3.4) are also unaffected by the uprated conditions, the revised parameters do not impact the pressurizer stress and fatigue analyses. It is therefore concluded that the pressurizer components meet the stress and fatigue analysis requirement of Section III of the ASME Code 1971 Edition, Summer 1971 Addenda for plant operation at the 1.5 percent uprated conditions.

3.6.2.8 Fuel Assembly

The Waterford 3 16 x 16 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. Consequently, the 1.5 percent uprate does not increase operating and transient loads such that they will adversely affect the fuel assembly functional requirements. Since the core plate motions for the seismic and LOCA evaluations are not affected by the uprated conditions, there is no impact on the fuel assembly seismic/LOCA structural evaluation.

Therefore, the fuel assembly structural integrity is not affected, and the normal operating, seismic and LOCA evaluations of the 16 x 16 fuel design for Waterford 3 are still applicable to the 1.5 percent uprate.

3.6.2.9 NSSS Piping and Pipe Whip

The reactor coolant main coolant loop piping system (including primary loop piping and pipe whip restraints, and tributary piping nozzles) was assessed for power uprate effects. It was concluded that these equipment designs remain acceptable and will continue to satisfy design basis requirements in accordance with applicable design basis criteria, which include the criteria associated with the original design basis mechanistic LOCA breaks, when considering the operating temperature, operating pressure, and flow rate effects resulting from the power uprate conditions.

In conclusion, Waterford 3 primary piping and tributary nozzles remain within allowable stress limits in accordance with ASME Section III, 1971 edition, including Addenda through Summer 1971. Furthermore, no piping or pipe restraint modifications are required as a result of the increased power level, because, per Section 3.6.1, the conservatively determined LOCA loads, for which the pipe restraint systems were designed, remain applicable for the uprate.

3.6.3 Effects of Operating Point Data Variations

The Appendix K Uprate operating point values shown in Section 3.3 represent a best estimate. In all probability, Appendix K Uprate operating point will move slightly over time, resulting in small changes in the operating point parameters.

Regardless of these small anticipated changes, particularly in the operating temperatures and the resulting ΔT, the structural AOR performed for the Waterford 3 RCS components will remain bounding. The following discussion is based on the fact that the AOR considered nominal T_{hot} and T_{cold} design values of 611 °F and 553 °F, respectively, with a resulting ΔT of 58 °F.

3.6.3.1 RCS Thermal Movements

The maximum thermal movements of various locations on the RCS (e.g., tributary nozzle ends) result from the change in RCS temperature from ambient conditions to operating conditions. Appendix K Uprate thermal movements will be enveloped by the AOR results, since AOR results are based on ambient to operating condition nominal temperature ranges that bound the temperature ranges associated with Appendix K Uprate. Furthermore, this conclusion will remain valid if the nominal values of T_{hot} and T_{cold} vary slightly during Appendix K Uprate, because there is sufficient margin between the Appendix K Uprate nominal T_{hot} value of 600.2 °F and the design T_{hot} of 611 °F, and between the Appendix K Uprate nominal T_{cold} of 545 °F and the design T_{cold} of 553 °F.

3.6.3.2 RCS Loads

RCS component nozzle and primary piping thermal expansion loads are directly affected by ΔT, the temperature difference between T_{hot} and T_{cold}. Given the same RCS configuration and operating temperatures that are generally the same, for example, lower ΔT values will result in lower piping and nozzle loads, which in turn will result in proportionally lower loads at intermediate component locations and at the component supports. This conclusion can be drawn because the general RCS characteristics of stiffness, mass and connectivity will not change for Appendix K Uprate, thus resulting in an overall RCS load distribution for Appendix K uprate conditions that will be very similar to the load distribution analyzed in the AOR.

The ΔT values associated with current and uprated conditions are both less than the ΔT value used in the AOR. Therefore, even though ΔT will increase slightly when going from the current to the uprated conditions, the AOR piping, component and component support thermal expansion loads will remain bounding, because they are associated with a higher value of ΔT.

The AOR design thermal transients remain bounding for the Appendix K uprate. Original design basis RCS seismic analysis results are negligibly affected by the uprate, because small changes in temperature have virtually no effect on the material properties of the structure, and therefore, on the manner in which the structure responds to a given set of input loads. Furthermore, Section 3.6.1 concludes that it is valid to base Appendix K LOCA evaluations on the original design basis events. Since the RCS structure will respond to the same design input loadings in essentially the same manner under Appendix K uprate conditions, the original design basis structural analysis results will remain valid.

3.6.3.3 RCS Stresses and Usage Factors

Since the original design transients, and the AOR NOP, seismic and LOCA structural analysis results remain bounding for the Appendix K uprate, the AOR Design, Emergency and Faulted condition load combinations used to calculate the stresses and fatigue usage factors of record also remain bounding. It is also noted that the ASME Code stress allowables used in the AOR are unaffected by small changes in operating temperatures, leading to the conclusion that the bounding stresses determined in the AOR will continue to remain below their corresponding Code allowables. Consequently, the structural integrity of the RCS components is further confirmed for small variations in the Appendix K uprate conditions, and the stress margins identified in the AOR calculations remain applicable.

3.6.4 Neutron Fluence

The existing projection of fast neutron fluence used in the reactor vessel design remains bounding for the uprated power conditions. This conclusion is based on a fluence evaluation that considered the changes to fuel management and the results of the last surveillance capsule evaluation (capsule W-97).

The RCS Pressure and Temperature (P/T) Limits in the Technical Specifications were based on the projected fluence at 20 effective full-power years (EFPYs). The P/T Limits are currently accepted for operation to 16 EFPYs. The reactor vessel will have completed approximately 14 EFPYs at the end-of-cycle 11. The next surveillance capsule is scheduled for removal and evaluation at that time. Changes will be made to the P/T Limits based on new fluence projections that include the effect of the power uprate conditions.

The power uprate from 3,390 MWt to 3,441 MWt may result in a slight increase (less than 2%) in the neutron flux and a negligible (less than 1%) increase in the 16 EFPYs fluence. Furthermore, a reduction in the original neutron fluence estimate was realized through the changes to fuel management such that the 20 EFPYs fluence used as the basis for the P/T Limits will bound the fluence at 16 EFPYs for the uprated power conditions. The reductions in fluence from fuel management will be measured as part of the next surveillance capsule evaluation at the end-of-cycle 11 and assessed for power uprate conditions to project reactor vessel fluence in future cycles.

3.7 NSSS/BOP FLUID SYSTEMS INTERFACE

The following BOP fluid systems were reviewed to assess compliance with NSSS/BOP interface guidelines at the revised design conditions shown previously in Section 3.3.

3.7.1 Main Steam System

The following subsections summarize the evaluation of the major steam system components relative to the revised operating conditions for the 1.5-percent power uprate. The major components of the main steam system (MS) include the steam generator main steam safety valves (MSSVs), the steam generator atmospheric steam dump valves (ADVs). Other major MSS components are the main steam isolation valves (MSIVs).

3.7.1.1 Main Steam Safety Relief Valves

The Main Steam Safety Relief Valves must have sufficient capacity so that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV Code) for any pressure transients anticipated to arise. Based on this requirement, a conservative criterion was applied that the valves should be sized to relieve 100 percent of the maximum calculated steam flow at an accumulation pressure not exceeding 110 percent of the design pressure.

Waterford 3 has twelve main steam safety relief valves (six on each main steam line) with a minimum total capacity of 15.83×10^6 lb/hr. These capacities are 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 revised design conditions. Therefore, based on the range of NSSS performance parameters for the uprating, the capacity of the installed MSSVs meets the sizing criterion.

3.7.1.2 Power Operated Atmospheric Dump Valves

There are two power operated atmospheric dump valves (one on each main steam line) which are installed upstream of the main steam isolation valves. The primary function of the Power Operated Atmospheric Dump Valves is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere whenever the main steam isolation valves are closed or when the condenser is not available. Under such circumstances, the ADVs, in conjunction with the emergency feedwater system (EFWS), permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the shutdown cooling system can be placed in service. During cooldown, the ADVs are either automatically or manually controlled. In automatic, each ADV proportional and integral (P&I)

controller compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

In the event of a tube rupture event in conjunction with loss of offsite power, the ADVs are used to cool down the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSV. RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere.

Each of the ADVs are 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 EFWS. The ADVs have a total design capacity of 1,600,000 lb/hr at 885 psig. For the revised design conditions, the ADV capacity is approximately 10.5 percent of the required maximum steam flow. Since the design capacity of the installed ADVs meets the sizing criterion, the values are adequately sized for the 1.5 percent uprated conditions.

3.7.1.3 Main Steam Isolation Valves

The MSIVs are located outside the containment and downstream of the MSSVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish this function, the MSIVs must be capable of an overall closure time of 8 seconds. These requirements are not impacted by the 1.5 percent power uprate because the present design is based on the full design pressure differential across the valves at a Rated Thermal Power of 102 % of 3,390 MWt.

3.7.2 Steam Bypass and Control Subsystem

The steam bypass and control system creates an artificial steam load by dumping-steam from ahead of the turbine valves to the main condenser. It is located downstream of the main steam isolation valves. Six parallel air-operated angle valves are connected to the main steam line header downstream of the main steam isolation valves. The sizing criterion is that the steam dump system (valves and pipe) be capable of discharging 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 main steam safety relief valves. If it is determined that the load rejection exceeds the capacity of the bypass valves, a demand signal is sent to the two redundant Reactor Power Cutback System comparators which will initiate the dropping of selected control element assemblies into the reactor core to reduce NSSS excess energy to a value within the capacity of the bypass valves.

Each of the six valves provide a steam dump capacity of 1.779 x 10^6 lb./hr at 915 psia inlet pressure. The total capacities provide steam dump capabilities of approximately 70 percent of the uprated steam flow (15.253 10^6 lb./hr, at a full-load steam generator pressure of 831.5 psia) versus the sizing criterion of 65 percent of rated steam flow. Therefore, the steam dump capacity is adequate for the 1.5 percent power uprate.

3.7.3 Feedwater System

The Feedwater system must automatically maintain steam generator water levels during steady-state and transient operations. The range of NSSS performance parameters results in a required feedwater volumetric flow increase, relative to present operation, of up to 1.8 percent during full-power operation. The higher feedwater flow has an impact on system pressure drop, which may increase slightly. The system has been evaluated to accommodate the system pressure drop for uprate.

The major components of the Feedwater System are the Main Feedwater Isolation Valves, the Main Feedwater Regulating Valves and the Main Feedwater Pumps.

Main Feedwater Isolation Valves / Main Feedwater Regulating Valves

The main feedwater isolation valves (MFIVs) are located just outside containment and downstream of the main feedwater regulating valves (MFRVs). The valves primary function is to isolate the steam generator and / or containment on receipt of a Main Steam Isolation Signal (MSIS). Isolation of feedwater flow is required to prevent containment overpressurization and excessive RCS cooldowns. Technical Specification Section 4.7.1.6 requires that they close within 5 seconds during accident conditions (see Reference 3.7.3-1). The accident and containment analyses that provided the bases for the Technical Specification were performed at 102 % of Rated Thermal Power or greater. The Main Feedwater Regulating Valves provide backup feedwater isolation by closing on the same Main Steam Isolation Signal as the MFIVs. The Main Feedwater Regulating Valves also close within 5 seconds of receipt of a MSIS. These requirements are not impacted by power uprate.

The feedwater flow for the proposed 1.5 percent power uprate is within the original design capability of both the Main Feedwater Isolation Valves and the Main Feedwater Regulating Valves. No modifications are therefore required for these valves due to the proposed uprate.

Main Feedwater Pumps

Two centrifugal type Steam Generator Feed Pumps are installed to feed the steam generators. Each pump is driven by a variable speed steam turbine and has a guaranteed capacity of 17,940 gpm at a discharge head of 2,150 ft. The operating speed range is from 3,500 rpm to 5,200 rpm. The two existing SG Feed Pumps have the capacity to support the proposed 1.5 percent power uprate without modification. The proposed uprate will require a slight increase in the operating speed. This increase will actually bring the pumps closer to their design speed and will thus slightly increase the efficiency of the pumps. No modifications are therefore required for these pumps due to the proposed uprate.

3.7.4 Emergency Feedwater System

The Emergency Feedwater System (EFW) provides cooling water to one or both steam generators for the purpose of removal of decay heat from the reactor coolant system (RCS) in response to any event causing low steam generator level coincident with the absence of a low pressure trip. Ordinarily EFW system actuation will be in response to any loss of main feedwater to the steam generators due to such initiating events as:

a. loss of main feedwater
b. loss of offsite power
c. station blackout
d. feedwater line break (inside or outside containment)
e. main steam line break (inside or outside containment)

The EFW system is not utilized during normal plant operating conditions.

The EFW system consists of two 50 percent capacity motor driven pumps and one 100 percent capacity steam turbine driven pump. Water is supplied from the Condensate Storage Pool (CSP) to connections at the two main feedwater lines. The steam turbine receives steam from either or both main steam lines (upstream of the main steam isolation valves).

The NRC Branch Technical Position (BTP) RSB 5-1 analysis was performed for the 8% Waterford 3 power uprate and the original accident analysis was performed at 102 % of rated thermal power. No change is therefore required to the EFW system for the proposed 1.5 percent uprate.

Condensate Storage Pool / Wet Cooling Tower Basin Requirements

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 Technical Specification required volume plus makeup from one WCT basin, ensures that sufficient water is available to cool the Reactor Coolant System to shutdown cooling entry conditions following any design basis accident.

The NRC Branch Technical Position (BTP) RSB 5-1 analysis was performed for the 8% Waterford 3 power uprate and the original accident analysis was performed at 102 % of rated thermal power. The proposed 1.5 percent power uprate will remain bounded by these analyses. No change is therefore required to the EFW water storage requirements.

3.7.5 Steam Generator Blowdown System

The steam generator blowdown system is used to control the chemical composition of the steam generator shell water to within the specified limits. The blowdown system also controls the buildup of solids in the steam generator water.

The blowdown flowrates 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, and 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, the rate of blowdown required to address dissolved solids should not 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. Therefore, the required blowdown to control secondary chemistry and particulates will not be significantly impacted by power uprate.

Since the inlet pressure to the steam generator 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. Therefore, the range of operating parameters revised for power uprate will not impact blowdown flow control.

Two of the COLSS secondary calorimetric inputs are blowdown mass flow rate and blowdown enthalpy. Both are dependent on steam generator pressure measurement. The operating pressure ranges of the steam generator were considered when the uncertainties of these parameters were determined.

3.7.6 Component Cooling Water / Auxiliary Component Cooling Water Systems

The Component Cooling Water (CCW) system is a closed cooling water system serving all reactor auxiliaries requiring cooling water. Heat is removed from the CCW system by Dry Cooling Towers (DCT) and by the Component Cooling Water Heat Exchangers. There are three CCW loops; a safety related essential loop, a nonessential seismically qualified loop and a nonessential non-seismic loop. The safety related essential loop consists of two independent trains (A and B). In the event of a Safety Injection Actuation Signal (SIAS) or a Containment Spray Actuation Signal (CSAS) the two safety related trains of the safety related essential loop are isolated from one another and from the nonessential loops.

The Auxiliary Component Cooling Water (ACCW) System cools the water in the CCW system via the Component Cooling Water Heat Exchangers and dissipates the heat to the atmosphere through the Wet Cooling Towers (WCT). The ACCW System is operated when the heat removal capacity of the dry cooling towers in the CCW system is not adequate to maintain the required CCW temperatures.

The major components of the CCW system are two CCW heat exchangers, three CCW pumps, two dry cooling towers, one surge tank and one chemical addition tank. The demineralized cooling water is pumped by the CCW pumps through the dry cooling towers and the tube side of the CCW heat exchangers. During normal operation, two CCW pumps are in operation and the third pump is on standby.

The ACCW system consists of two 100 percent capacity, safety related, independent trains. Each train includes a pump, an evaporative wet type mechanical draft cooling tower and a control valve. Water is supplied to the Wet Cooling Towers from the Wet Cooling Tower Basins.

For normal plant operation, the function of the CCW and ACCW Systems is to remove heat from mechanical components and heat exchangers (reactor auxiliaries).

The safety related function of the CCW and ACCW Systems is to:

1. Remove heat from the containment and reject the heat via the cooling towers to the atmosphere following a Loss of Coolant Accident (LOCA), a secondary system pipe break or a Main Steam Line Break (MSLB) inside containment.

2. Supply component cooling water to Containment Fan Coolers, Emergency Diesel Generators, Shutdown Cooling Heat Exchangers, Essential Chillers and Engineered Safety Features pumps.

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

3.8 BALANCE-OF-PLANT SYSTEMS

3.8.1 Heat Balance

The original design of the Waterford 3 secondary side components was done for a rated thermal power of 3,559 MWt which bounds the proposed uprate to 3,441 MWt. Balance of Plant conditions for a RTP of 3,559 MWt are shown on the Waterford 3

Valves Wide Open Heat Balance (Reference 3.8.1-1). New Balance of Plant (BOP) calculations were generated to determine the operating conditions for the Waterford 3 secondary side components for the proposed 1.5 percent power uprate. The new operating conditions were then compared to the conditions as shown on the original Waterford 3 secondary side design Valves Wide Open (VWO) heat balance. The secondary side system temperatures and pressures for the proposed 1.5 percent power uprate remain bounded by the original design pressures and temperatures. The results of the comparison are described below.

3.8.2 Feedwater System

The feedwater system supplies heated feedwater to the steam generators under all load conditions maintaining level within the programmed band. Level is maintained by positioning the feedwater control valve in the feedwater line to each steam generator.

For the power uprate, the feedwater flow rate will increase slightly for each unit but, will remain below system design capabilities. Feedwater flow rate and velocity through the feedwater heaters will also increase accordingly as a result of the power uprate but remain within the design of the components.

3.8.3 Feedwater Heater System

The primary function of the Feedwater Heater system is to supply preheated condensate, via the feedwater heater trains, to the suction of the steam generator feedwater pumps. The feedwater heater system pressure, temperature, and flow rate will change slightly at the uprate power level. However, these parameters will still remain below the system and component design conditions. The condensate pumps have sufficient margin to continue to satisfy feed pump flow rate and net positive suction head requirements at the uprated conditions.

3.8.4 Condenser

Steam flow to each condenser will increase as a result of the power uprate. However, the uprate conditions are bounded by the condenser design.

3.8.5 Extraction Steam System

The extraction steam system transmits steam from the high- and low-pressure main turbines to the shellside of the feedwater heaters for feedwater heating. During normal operation, steam from the high-pressure turbine is used to heat feedwater flowing through the first and second point heaters, and steam from the low-pressure turbines is used to heat feedwater flowing through the third, fourth, fifth, and sixth point heaters.

Implementation of uprate will yield greater extraction steam pressures, temperatures, and, in most cases, flows as indicated on the uprate heat balances. However, the uprate extraction steam conditions are bounded by the extraction steam system design. The flow velocities at current and uprate conditions are within equipment design limits and will not appreciably increase flow-accelerated corrosion relative to existing levels. Additionally, the extraction steam system is capable of precluding turbine water induction and minimizing the effects of flashing extraction steam on turbine overspeed at uprate conditions.

3.8.6 Heater Drains System

The heater drain system (HDS) and associated equipment were evaluated to ensure the ability of the system to function under power uprate conditions. HDS design parameters were reviewed and compared against power uprate conditions to determine that acceptable design margin exists for operation at uprate conditions.

Pressures and temperatures associated with the power uprate will remain bounded by the existing designs of the HDS and its components. HDS components will remain capable of passing additional flow rate associated with the power uprate conditions and component velocities will not exceed accepted maximum values.

3.8.7 Circulating Water System

The Circulating Water System (CWS) is an open-loop system that provides cooling water for the main condenser of the turbine generator unit. The cooling water is taken from and discharged to the Mississippi River. The total design circulating water flow rate is approximately 1,000,000 gpm.

The CWS system flow will remain essentially unchanged following power uprate. The increased levels of rejected heat, from an increase in turbine exhaust flow, will increase the CWS outlet temperature by less than 0.5 °F. The heat load under power uprate conditions will result in a slight backpressure increase in the condenser. However, the increased backpressure will remain within acceptable limits. The increase in outlet temperature, due to the increased heat load, is bounded by the CWS system design and can be accommodated by the system. No modifications to the CWS or its components are required for a power uprate.

3.8.8 Turbine Generator

The capability of the Main Turbine to perform at the proposed uprated power conditions was evaluated. The review included the throttle valves, high-pressure and low-pressure turbines, as well as associated auxiliary equipment including moisture separator reheaters (MSRs) and relief valves. All main turbine components were determined to have sufficient margin to enable operation at the uprated power conditions without requiring equipment modifications.

The existing turbine missile analysis remains valid for the proposed 1.5 percent power uprate since:

1. The kinetic energy of the rotating turbine components will remain unchanged (turbine rpm is not changing and no physical changes are being made to the turbine internal components).
2. No physical changes are being made to the turbine casings.
3. The secondary side system temperatures and pressures for the proposed 1.5 percent power uprate remain bounded by the original design pressures and temperatures.
4. No physical changes are being made to the containment building or reactor auxiliary building.
5. No physical changes are being made to components within the containment building or reactor auxiliary building.

No changes are therefore required for the Turbine Generator for the proposed power uprate to 3,441 MWt.

3.8.9 Turbine Component Cooling Water System

The Turbine Component Cooling Water (TCCW) System provides an intermediate cooling loop for removing heat from the turbine plant auxiliary systems and transferring it to the Circulating Water System. The system removes heat from designated non-safety-related turbine plant components. The heat is then transferred to the circulating water via the two turbine component cooling water heat exchangers. Since the original design of the Turbine Component Cooling Water System was based on a thermal power of 3,559 MWT no changes are required for the proposed 1.5 percent power uprate.

3.8.10 Balance Of Plant Piping, Pipe Supports and Pipe Whip

Balance of Plant (BOP) piping system was evaluated for the proposed 1.5 percent power up rate condition. The evaluations performed have concluded that these piping systems remain acceptable and will continue to satisfy 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. The design pressures and temperatures of the BOP piping, for the proposed power uprate, remain unchanged from the original design. Waterford 3 piping and related support systems remain within allowable stress limits in accordance with ASME Section III, 1971 edition, including addenda through winter 1972 for class 2 and 3 piping and ANSI B31.1 1973 edition as appropriate. The evaluations also concluded that no piping or pipe support modifications are required due to the increased power level.

The evaluation also included the effects of the power uprate on pipe break, jet and whip restraints and transients due to fast valve closure of Feed Water Isolation

Valves. No new postulated pipe break locations were identified in high energy piping. Jet impingement loading and pipe whip forces of the original design remain bounded by the power uprate condition.

3.9 ELECTRICAL SYSTEMS

3.9.1 Generator and Support Systems

The electrical systems associated with the turbine auxiliary systems are not affected by the uprate.

The generator has a design rating of 1333.2 MVA at 25 kV 60 Hz when operating with 60 psig hydrogen pressure at a 0.9 lagging power factor (1200 MW & 30789 amperes).

The generator shall be operated to produce power output (VAR, VA, WATT) within its Generator Capability Curve. No modification to the auxiliary equipment is required.

A review of applicable calculations identified no need for any changes to equipment protective relay settings for the generator.

To deliver electrical power from the generator to the transmission system, the unit is equipped with a main isolated phase (isophase) bus and splits into two secondary isophase busses, 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 1333.2 MVA.

The isophase bus main section is rated at 33,000 amps. The bus conductor is rated for a temperature of 65 °C rise forced cool. These temperature ratings will permit a total load of 1428.9 MVA. Each of the secondary isophase bus is rated at 15,000 amps at 65 °C rise forced cool and has an increase rating of 20,000 amps at 65 °C rise emergency cooling. The isophase bus temperature ratings are well in excess of the 101.5% generator output. The isophase bus will support the power increase with no modifications.

Each main transformer is rated at 600 MVA. 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 form 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.

Standard design practice at Entergy requires that switchyard equipment meet or exceed the rated capacity of the main generator. The Waterford 3 switchyard will accept the additional load without the need for any hardware modifications.

In summary, the turbine/generator and major electrical components extending from the isophase bus to the switchyard have adequate design margin to accept the additional power anticipated by the 1.5 percent uprate.

3.9.2 Onsite Distribution System

The onsite AC power system includes a class 1E system and a non-class 1E system. The onsite AC power system consists of Unit 3 main turbine-generator, two unit auxiliary transformers, two emergency diesel generators, and AC distribution system with nominal ratings of 6.9 kV, 4.16 kV, 480 volts, and 208/120 volts. 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.

3.9.2.1 Non-Class 1E AC System

The non-Class 1E AC system distributes power at 6.9 kV, 4.16 kV, 480 volts, and 208/120 volts 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 the 230 kV preferred offsite power source through the secondaries of the startup transformers consisting of dual windings (230kV to 6.9kV and 230kV to 4.16kV).

The 4.16 kV non-Class 1E auxiliary system is comprised of four buses (2A, 2B, 4A, and 4B). 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 480V level. The 6.9kV buses power the circulating water pumps and condensate pumps for the secondary system. The secondary side was originally designed for NSSS rating of 3559 MWT. Hence the large pumps and motors on the 6.9kV and 4.16kV buses are adequately sized. The cables and protective relaying is based on nominal rating of the motors and these are not affected.

The reactor coolant pumps are fed from non-Class 1E 6.9 kV auxiliary system buses 1A and 1B. The 1.5 percent increase in thermal power does not affect the T_{cold} temperature and hence, the RCPs loading is not affected. The cables and protective relaying for the RCPs are not affected by the power uprate.

The non 1E startup transformers are capable of supplying all of the startup or normal plant operating loads of the unit or the engineered safety feature (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.

3.9.2.2 Class 1E AC System

The Class 1E AC system consists of two separate trains and distributes power at 4.16kV, 480 volts, and 120 volts 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 (3A and 3B). 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 emergency diesel generator. In the event of a loss of offsite power (LOOP), the Class 1E AC system will be powered from the emergency diesel generators.

The large 4.16kV loads consist of HPSI, LPSI, Containment Spray, Emergency Feedwater, Component Cooling Water, and auxiliary Component Cooling Water pumps. As referenced in section 3.10 of this submittal, the power uprate of 1.5 percent is within the design bases of the original design of the plant for operation at 102% plant rating. The large pumps and motors are sized for safe shutdown following a design basis event with plant at an initial power of 102%. 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.

The existing emergency diesel generators (EDG) are rated at 4.4 MW with 10% 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 nominal rating of the EDGs to accommodate any minor variations in electrical loads. The minor increase in the decay heat load, post accident is considered within the error margin for the time based fuel oil consumption calculation.

3.9.2.3 120 Volt AC and 125 Volt DC Systems

The DC system is made of four trains. Each train has a battery, two battery chargers, and power distribution panels (PDPs). The chargers convert 480 VAC to DC using silicon controlled rectifiers and silicon diodes.

The major DC loads on the 1E battery systems are the static uninterruptible power supplies which power 120V AC system and control power for switchgear and critical valves. The system consists of nine static uninterruptible power supplies (SUPS). Six of the SUPS are safety related and the remaining three SUPS are non-safety related. The six safety related SUPS are SUPS MA, MB, MC, MD, A and B. Four of these SUPS, MA, MB, MC and MD, are the Instrument SUPS since the primary loads of these SUPS are measuring devices. The two remaining safety related SUPS, A and B, are referred to as the Vital SUPS since their loads are required for plant operation. The change in power requirements for these loads 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 CheckPlus instrumentation has been evaluated to be acceptable for the respective Panels.

The major loads on the non 1E DC system powered by the TGB battery are emergency lube oil and seal pumps. These pumps are required for turbine coastdown upon reactor trip and loss of offsite power 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 208VAC and 120VAC 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 Low Voltage Distribution 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 120/208V system.

3.9.2.4 Onsite Distribution System Review

The impact of potential increases in brake horsepower loads on non-safety related pumps (i.e., condensate pumps, heater drain pumps, circulating water pumps, etc.) due to the 1.5 percent. power uprate have been determined to be insignificant. Based on review of the onsite equipment rating, sizing criteria, existing loading, and margins, the electrical equipment powered by the onsite distribution system remains within their respective ratings. Thus, the onsite distribution system is not affected by the uprate.

3.9.3 Grid Stability

Entergy performs the grid system analysis to support the proposed power increase. The grid system analysis was performed for a bounding uprate of 2% increase assuming a bounding gross generator output of 1170 (1150 + 2%) 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 the review of the current analysis, current grid reliability and stability are not impacted and Waterford 3 continues to be in conformance with the General Design Criterion 17 for the power uprated electrical conditions.

3.10 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT EVALUATION

3.10.1 Plant Protection System Setpoints

As discussed in section 3.5.8, the Waterford 3 PPS is comprised of an Engineered Safety Features System (ESFAS) and a Reactor Protection System (RPS). The ESFAS consists of sensors, logic and other equipment necessary to monitor selected NSSS and containment conditions in order to generate signals to actuate the ESF and ESF support systems. The Reactor Protective System (RPS) is that portion of the PPS which generates signals that actuate reactor trip.

PPS setpoints are established by adding instrument channel uncertainties to the instrument channel analysis limit. The instrument channel analysis limit is based on assumptions made that support the Waterford 3 safety analyses as documented in Chapter 15 of the FSAR. The discussion below addresses the impact of this uprate on the analysis limits for each of the PPS setpoints.

The instrument channel uncertainties are the combination of error effects that are inherent with instrument channel components, calibration acceptance limits, calibration material and test equipment (M&TE), and process measurement effects (PME). This uprate does not modify or change PPS channel components. Consequently, error attributes such as reference accuracy, drift effects, ambient temperature effects, etc are not affected by this power uprate. This uprate does not change the calibration acceptance tolerances or accuracies of the M&TE used for calibration; uncertainties due to calibration practices are not affected by this power uprate. Also, the method and values used to derive the periodic test error (PTE) and allowable values as described in the Technical Specification LSSS (RPS and ESFAS) bases are not affected by the uprate.

Uncertainty attributes that are potentially affected by this power uprate are process measurement effects. The affect of this power uprate on PME is also discussed below.

Linear Power Trip Level - High

The linear power level high trip is generated by two out of four excore neutron detection channels to initiate a reactor trip. The function of this trip is to provide reactor core protection against reactivity excursions. This trip function is not explicitly credited in the accident analyses. The analysis limit in the technical specifications is not changed.

These channels deviate from true reactor power as a result of normal core burn-up over a cycle. These channels are adjusted to match COLSS power on a regular basis. The power uprate will have no impact on the accuracies of these channels since they are periodically reconciled to CPC or COLSS power indication.

The detectors will be able to be re-scaled to detect the increased neutron flux resulting from this uprate. The setpoint, as listed in the Technical Specifications, will not be affected by power uprate.

Logarithmic Power Level - High

The Logarithmic Power Level High trip is generated by two out of four excore logarithmic neutron detection channels to initiate a reactor trip. The function of this trip is to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. (e.g., CEA Withdrawal (CEAW) from subcritical) (Table 4-2, UFSAR Section 15.4.1.1). The impact of power uprate on this trip is the increase in the analysis limit of the trip setpoint from 4.4% of 3,390 MWt to 4.4% of 3,441 MWt. An examination of the current analysis of record showed that this would result in a change in trip time of less than 3/1000 of a second, which is negligible. A reactor trip is initiated by the Logarithmic Power Level - High trip at a thermal power level as measured by the excore logarithmic power channels of less than or equal to 0.257% of the new rated thermal power of 3,441 MWt. This trip may be bypassed by the operator when the thermal power level is above 10^{-4} % of 3,441 MWt (the new rated thermal power). This bypass is automatically removed when the thermal power decreases below 10^{-4} % of 3,441 MWt.

The uncertainty of this measurement channel is therefore unchanged by power uprate. The detectors will be able to be re-scaled to detect the increased neutron flux resulting from this uprate. The re-scaling will not affect detector uncertainties since the detector channel uncertainties are provided as a percent of equivalent linear full scale, and are directly proportional to the power uprate change. The setpoint of this trip function is not changed.

Pressurizer Pressure - High

The pressurizer pressure high trip is generated by two out of four narrow range pressurizer pressure channels to initiate a reactor trip. The function of this trip, in conjunction with the pressurizer safety valves and main steam safety valves is to provide Reactor Coolant System protection against overpressurization. This trip's setpoint is at less than or equal to 2350 psia which is below the nominal lift setting of 2500 psia for the pressurizer safety valves.

Based on section 3.3, the operating pressure of the pressurizer has not changed. Process measurement effects and other uncertainties associated with pressurizer pressure are unaffected. This trip setpoint is not affected by this uprate.

Pressurizer Pressure - Low

The pressurizer pressure low trip is generated by two out of four wide range pressurizer pressure channels to initiate a reactor trip, an ESFAS safety injection actuation signal (SIAS) and an ESFAS containment isolation actuation signal (CIAS). On high-high containment pressure, low pressurizer pressure also initiates a containment spray actuation signal (CSAS). During normal operation, this trip's setpoint is set at greater than or equal to 1684 psia.

Based on section 3.3, the operating pressure of the pressurizer has not changed. Process measurement effects and other uncertainties associated with pressurizer pressure are unaffected. This trip setpoint is not affected by this uprate.

This trip's setpoint may be manually decreased to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns. The margin between the pressurizer pressure and this trip's setpoint is maintained during shutdowns at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. Based on the discussion above, this function remains unaffected by the uprate.

Containment Pressure - High

The containment pressure high trip is generated by two out of four containment pressure channels to initiate a reactor trip, an ESFAS safety injection actuation signal (SIAS), an ESFAS containment isolation actuation signal (CIAS), and an ESFAS main steam isolation signal (MSIS). The setpoint for this trip is less than or equal to 17.1 psia.

Based on the discussion in section 3.5.5, the basis for the containment cooling limits are not changed by this uprate. Based on section 3.5.5, the operating pressure of the containment has not changed. Process measurement effects and other uncertainties associated with containment pressure are unaffected. This trip setpoint is not affected or changed by the uprate.

Containment Pressure – High - High

The containment pressure high-high trip is generated by two out of four containment pressure channels to initiate an ESFAS containment spray actuation signal. The setpoint is set at less than or equal to 17.7 psia.

Based on the discussion in section 3.5.5, the basis for the containment cooling limits are not changed by this uprate. Based on section 3.5.5, the operating pressure of the containment has not changed. Process measurement effects and other uncertainties associated with containment pressure are unaffected. This trip setpoint is not affected or changed by the uprate.

Steam Generator Pressure - Low

The steam generator pressure low trip is generated by two out of four steam generator pressure channels to initiate a reactor trip and an ESFAS main steam isolation signal (MSIS).

The trip setpoint of 764 psia is sufficiently below the uprate full load operating point of approximately 831 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. The uprate operating pressure is a lower steam generator pressure. The channels that initiate this trip measure pressure directly and are not subject to pressure related process measurement effects or static pressure span or zero shifts. This uprate will not affect this PPS setpoint.

This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns. The margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi. This setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached. Based on the discussion above, this function remains unaffected by the uprate.

Steam Generator/Steam Generator ΔP - High

The high ΔP between steam generators high trip is generated by two out of four steam generator pressure channels in conjunction with two out of four low level channels to initiate an ESFAS emergency feedwater actuation signal (EFAS).

As discussed above, the uprate operating pressure of the steam generators is lower, but the channels that actuate this trip measure steam generator pressure directly and are not subject to pressure related process measurement effects or static pressure span or zero shifts. This uprate will not affect the current PPS setpoint of 123 psid.

Steam Generator Level - Low

The low steam generator level trip is generated by two out of four narrow range steam generator level channels in conjunction with two out of four high high steam generator/steam generator ΔP channels to initiate a reactor trip and an ESFAS EFAS. This function provides protection against events involving a mismatch between steam and feedwater flow. These may be due to a steam or feed line pipe break or other increased steam flow or decreased feed flow events. A large feedwater line break event inside containment results in a potential loss of a steam generator heat sink.

The current setpoint of this trip is 27.4% narrow range level. From the discussion in section 3.3, steam generator pressure is expected to drop slightly as a result of this uprate. A drop in SG pressure causes indicated steam generator level to increase relative to the actual level. This effect potentially actuates the trip setpoint at lower actual level. However, the magnitude of this change added to existing uncertainties remains well within the available measurement margin. Therefore the change in steam generator pressure resulting from this uprate will have no effect on the steam generator low level trip setpoint.

Steam Generator Level - High

The high steam generator level trip is generated by two out of four narrow range steam generator level channels to initiate a reactor trip. This function protects the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. It's functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

The current setpoint of this trip is 87.7% narrow range level. From the discussion in section 3.3, steam generator pressure is expected to drop slightly as a result of this uprate. However the drop in steam generator pressure is not expected to cause the net high level trip setpoint to decrease below the normal operating level of the steam generators. This uprate will not result in any additional challenges to the PPS by increasing the chances of spurious actuation in normal operating bands of plant equipment.

Steam Generator Wide Range Level - Low

The low steam generator wide range level setpoint is generated by one out of two wide range steam generator level channels in conjunction with an EFAS. This function provides protection against events involving a mismatch between steam and feedwater flow. These may be due to a steam or feed line pipe break or other increased steam flow or decreased feed flow events.

The current setpoint of this trip is 36.3% wide range level. The effect discussed above for the narrow range low level trip under normal power uprate conditions does not apply to this particular setpoint because the uncertainty analysis assumes steam generator pressure has declined to as low as 134 psia under accident conditions. The drop in steam generator pressure resulting from the uprate has no effect on this setpoint.

Local Power Density - High

The local power density (LPD) high trip is initiated by two out of four of the core protection calculators (CPCs). This function prevents the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system
b. Radial peaking factors from the position measurement for the CEAs
c. Delta T power from reactor coolant temperatures and coolant flow measurements.

The local power density (LPD) incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit..

As stated in section 3.5, the increase in rated thermal power (RTP) to 3,441 MWt will require changes to the CPC constants that set the core average heat flux and core average linear heat rate for the various algorithms. This affects the VOPT setpoints, which are discussed in Section 3.10.4.1.

DNBR - Low

The DNBR Low trip is initiated by two out of four of the CPCs. This function is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system
b. Reactor Coolant System pressure from pressurizer pressure measurement
c. Primary calorimetric (Delta T) power from cold and hot leg temperatures and coolant flow measurements
d. Radial peaking factors from the position measurement for the CEAs
e. Reactor coolant mass flow rate from reactor coolant pump speed
f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than the fuel design limit such that the decrease in actual core DNBR after the trip will not result in a violation of the DNBR Safety Limit of 1.26. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modeling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

As discussed in section 3.3, the uprate will not cause any of the parameters described above to exceed or challenge the CPC limits.

As stated in section 3.5, the increase in rated thermal power (RTP) to 3,441 MWt will require changes to the CPC constants that set the core average heat flux and core average linear heat rate for the various algorithms.

Reactor Coolant Flow - Low

The reactor coolant flow low trip is initiated by two out of four tube-side steam generator ΔP channels. This function trips the reactor and provides protection against a reactor coolant loss of flow type event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a nominal setpoint of 19.0 psid. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

After the uprate, the reactor coolant pumps will be pumping the same volumetric flow. The net ΔP across the steam generators will not change. The analysis limit and measurement margins will remain unaffected by the power uprate. The uprate has negligible impact on RCS nominal pressure. The uncertainties of the measurement channels as a result of process measurement effects or static pressure span and zero shifts due to negligible RCS pressure changes are not affected. The setpoint for this PPS function is therefore not affected by this power uprate.

Refueling Water Storage Pool (RWSP) Level - Low

The RWSP low level setpoint is initiated by two out of four RWSP level measurement channels. This function initiates a recirculation actuation signal (RAS). The RAS automatically aligns the ECCS to draw water collected in the containment SIS sump through the shutdown heat exchangers then back into containment for RCS and containment cooling. The RAS actuates when RWSP level decreases to 10%.

The ECCS will be capable of performing its design function after the uprate with no changes. RWSP level limits are not affected. Also, none of the process parameters are changed, so the instrument channel uncertainties are not affected.

Loss of Electrical Power

Loss of electrical power resulting from a loss of offsite power is detected by loss of voltage and degraded voltage relays. This function starts the emergency diesel generators to support safe plant shutdown under normal and accident conditions.

From section 3.9 below, the voltage on the buses that are monitored by these relays (480 V and 4.16 kV) is not changed as a result of this uprate. The relay setpoints are therefore not affected by this uprate.

3.10.2 Emergency Core Cooling System Performance

The Waterford 3 ECCS performance analysis consists of three analyses, namely, the Large Break Loss-of-Coolant Accident (LBLOCA), Small Break Loss-of-Coolant Accident (SBLOCA), and post-LOCA Long Term Cooling (LTC) analyses. The LBLOCA and LTC analyses are performed at a core power level of 3,458 MWt; the SBLOCA analysis is performed at a core power level of 3,478 MWt. Consistent with the original requirement of Paragraph I.A of Appendix K to 10 CFR 50, these values are equal to or greater than 102% of the current licensed core power level of 3,390 MWt.

As allowed by the recent revision to Paragraph I.A of Appendix K, this operating license/Technical Specification change 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%. With these proposed revisions, the value for the licensed core power level plus the maximum power measurement uncertainty remains 3,458 MWt. Since the Waterford 3 ECCS performance analysis was performed at a core power level that is greater than or equal to 3,458 MWt, it complies with the revised requirement of Paragraph I.A of Appendix K for the proposed values for the licensed core power level and power measurement uncertainty.

A review of the plant data that are impacted by the proposed revisions (for example, the nominal operating point) concluded that there are no changes to the data used in the Waterford 3 ECCS performance analysis.

Consequently, since there is no change to the core power level used in the ECCS performance analysis and there are no changes to any other inputs to the analysis as a consequence of the proposed revisions, there are no changes to the Peak Cladding Temperature or any other result of the Waterford 3 ECCS performance analysis.

The Waterford 3 ECCS performance analysis is performed with the Westinghouse ECCS performance evaluation models for Combustion Engineering designed PWRs. The topical reports that describe the evaluation models (References 3.10.2-1, 3.10.2-2, and 3.10.2-3) explicitly state that 102% of the licensed core power level will be used in the analyses. For example, Section III.A of CENPD-132P states that *"The reactor will be assumed to be operating at a power level of 102 percent of the maximum licensed power."* As described in Section 2.0 of this Technical Specification change request, EOI proposes that the Westinghouse ECCS performance evaluation models for Combustion Engineering designed PWRs be accepted for use with the proposed license amendment changes described herein without revisions to the evaluation model topical reports to address the recent revision to Paragraph I.A of Appendix K.

3.10.3 Non-LOCA/Transient Analyses

The Waterford 3 Non LOCA Transient Analyses is based on the eight by three matrix specified in Reference 3.10.3-1. Initiating events are placed in one of the eight categories of process variable perturbation. The frequency of each incident was estimated, and each incident was placed in one of three frequency categories specified in Reference 3.10.3-1. The initial power level assumed for the Non LOCA events included a 2% power uncertainty for those events in which higher power produced more adverse results.

Additionally, the current radiological consequences calculation, which forms a reload design limit on the extent of predicted fuel cladding failures during limiting faults, is based on the steam release following operation at a core power in excess of 102%. The inventory of fission products available for release upon clad failure has been verified to be applicable to operation at 3,441 MWt (see Section 3.12). Thus the overall radiological consequences are not adversely impacted by the increase of rated thermal power to 3,441 MWt.

As allowed by the recent revision to Paragraph I.A of Appendix K, it is proposed to increase the licensed core power level by 1.5 percent to 3,441 MWt. Consequently, there is no increase in the total core power level for the Waterford 3 Non LOCA transient analyses. In addition, a review of the plant design data used in the Waterford 3 Non LOCA transient analyses concluded that there are no adverse changes to any plant design data used in the analysis as a result of the proposed changes to the licensed core power level and power measurement uncertainty. Consequently, since there is no change to the total core power level used in the Non LOCA transient analysis or to any other inputs to the analysis as a result of the proposed changes to the licensed core power level and power measurement uncertainty, there are adverse changes in the docketed results of the Non LOCA transient analysis.

Following is a brief discussion of the impact of power uprate on Reactor Protection System trip setpoints. The impact of the power uprate on the dynamics of the transients is shown in Table 3.10.3-1.

3.10.3.1 Other Trip Setpoints

This power uprate will have an impact on the trip setpoints which are based on a percentage of the rated thermal power (RTP). The reactor trips that are based on a percentage of the RTP are

1. High Log Power Trip (see Section 3.10.1)
2. High Linear Power Trip (see Section 3.10.1)
3. Core Protection Calculator System (CPCS) Variable Overpower Trips (VOPT)
   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)

CPCS VOPT SPVMIN

The CPCS SPVMIN, the floor for the VOPT, is used as mitigating action against transients starting from a low power state (e.g., CEAW from Hot Zero Power (HZP)) (Table 4-2, UFSAR Section 15.4.1.2). Currently the floor of the VOPT, SPVMIN, is set at 30% of 3,990 MWt. To maintain the credited reactor trip at the same absolute power level, SPVMIN will be reduced by the ratio of the new and old Rated Thermal Power definitions. Thus, for operation at a Rated Thermal Power of 3,441 MWt, SPVMIN will have a setpoint of 29.6% of 3,441 MWt.

CPCS VOPT SPVMAX

The maximum value of the CPCS VOPT, SPVMAX, is a high power trip setpoint. Currently that value is set to 110.0% of 3,390 MWt. To maintain the same relationship between the initial conditions and the trip setpoint in terms of absolute power changes, the setpoint SPVMAX will be reduced by the ratio of the new and old Rated Thermal Power definitions. Thus, for operation at a Rated Thermal Power of 3,441 MWth, SPVMAX will have a setpoint of 108.3% of 3,441 MWt.

CPCS VOPT "Rate of Change" (SUPMAX, SDNMAX)

At steady state power conditions the trip setpoint is set 8% 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%/Minute. To maintain the same relationship between the transient conditions and the trip setpoint in terms of absolute power changes, the setpoint SUPMAX is being reduced by the ratio of the new and old Rated

Thermal Power definitions. Thus, for operation at a Rated Thermal Power of 3,441 MWt, SUPMAX will be reduced to 1.97% 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.

CPCS VOPT "Offset"

The VOPT setpoint is set by an offset above the steady state power level. This offset, DELSPV is currently set to 8% 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 to SPVMAX. To maintain the same relationship between the initial conditions and the trip setpoint in terms of absolute power changes, the setpoint DELSPV is being reduced by the ratio of the new and old Rated Thermal Power definitions. Thus, for operation at a Rated Thermal Power of 3,441 MWt DELSPV will be reduced to 7.8% of 3,441 MWt.

3.10.3.2 Steam Generator Tube Plugging

The tube plugging assumptions used in the current accident analyses performed for Waterford 3 is based on a range of tubes plugged, from 0 tubes plugged (clean Steam Generator) up to 500 tubes plugged per SG. The power uprate has no direct impact on the tube plugging assumptions used for the UFSAR Chapter 15 Analyses and as seen below, events which are limiting at either extreme of the plugged tube spectrum have already been analyzed at the same "rated thermal power plus uncertainty" which will exist following the power uprate.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.1 Increase in Heat Removal by the Secondary System			
15.1.1.1	Decrease in Feedwater Temperature	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26, Ref. 3.10.2-2, TS 2.1.1.1 and Peak LHR ≤ 21 kW/ft, Ref.3.10.2-2, TS 2.1.1.2)	Not analyzed since all criteria are bounded by Increased Main Steam Flow (UFSAR Section 15.1.1.3).
15.1.1.2	Increase in Feedwater Flow	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26 and Peak LHR ≤ 21 kW/ft)	Not analyzed since all criteria are bounded by Increased Main Steam Flow (UFSAR Section 15.1.1.3).

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.1.1.3	Increased Main Steam Flow	Peak RCS Pressure $\leq$ 110% of Design Peak Secondary Pressure $\leq$ 110% of Design No Fuel Failure (Minimum DNBR $\geq$ 1.26 and Peak LHR $\leq$ 21 kW/ft)	Peak Pressure criteria are not challenged for this event. CPCS filters are set to ensure DNBR trip to preclude fuel failure. The filter verification is impacted by the rate of change of Tcold and is not impacted by the power uprate. Therefore, the power uprate has no adverse impact on any of the criteria for this event.
15.1.1.4	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve	Peak RCS Pressure $\leq$ 110% of Design Peak Secondary Pressure $\leq$ 110% of Design No Fuel Failure (Minimum DNBR $\geq$ 1.26 and Peak LHR $\leq$ 21 kW/ft)	Peak Pressure and Fuel Performance criteria are bounded by Increased Main Steam Flow (UFSAR Section 15.1.1.3). The most adverse offsite dose consequence for this event occurs at Hot Zero Power (HZP) and no trip is credited for this event. Therefore, the power uprate has no impact on any of the acceptance criteria for this event The radiological consequences are bounded by the IOSGADV with Single Active Failure (SF) (UFSAR Section 15.1.2.4).

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.1.2.1	Decrease in Feedwater Temperature With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design Maintain coolable geometry Offsite Doses well within 10CFR100 guidelines	Not analyzed since all criteria are bounded by Increased Main Steam Flow with Single Active Failure (UFSAR Section 15.1.2.3).
15.1.2.2	Increase in Feedwater Flow With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design Maintain coolable geometry	Not analyzed since all criteria are bounded by Increased Main Steam Flow with Single Active Failure (UFSAR Section 15.1.2.3).

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.1.2.3	Increased Main Steam Flow With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design Maintain coolable geometry Offsite Doses well within 10CFR100 guidelines	Peak Pressure criteria are not challenged for this event. The minimum DNBR case for this event is modeled as an initial event which degrades the initially preserved thermal margin followed by a 4 pump Loss of Flow from SAFDL conditions. The flow input into the CPCS which generate the reactor trip is unaffected by the power uprate. The CPCS calculation of SAFDL conditions which ensures the thermal margin conditions at the start of the flow coastdown will be valid at power uprate conditions. Therefore, the power uprate has no impact on any of the acceptance criteria for this event. As discussed in Section 3.12, all radiological consequences continue to meet the acceptance criteria.
15.1.2.4	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design Maintain coolable geometry Offsite Doses well within 10CFR100 guidelines	Peak Pressure and Fuel Performance criteria are bounded by Increased Main Steam Flow with Single Active Failure (UFSAR Section 15.1.2.3). The most adverse offsite dose consequence for this event occurs at HZP and there is no trip credited for this event. Therefore, the power uprate has no impact on any of the acceptance criteria for this event. As discussed in Section 3.12, all radiological consequences continue to meet the acceptance criteria.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.1.3.1	Steam System Piping Failures (Post - trip return to power)	Maintain coolable geometry Offsite Doses a small fraction of 10CFR100 guidelines (with no iodine spike). Offsite Doses within 10CFR100 guidelines (with pre-existing iodine spike).	The most adverse consequence for this event occurs at HZP. The RPS trip is based on the Low SG Pressure trip which is not impacted by the power uprate. The HFP cases have already been analyzed at 102% of 3,390 MWth. Therefore, the power uprate has no impact on any of the acceptance criteria for this event. As discussed in Section 3.12, all radiological consequences continue to meet the acceptance criteria.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.1.3.2	Steam System Piping Failures (Lower Mode ARI Return-to-Power)	Maintain coolable geometry Offsite Doses within 10CFR100 guidelines.	As this is an analysis initiated from subcritical conditions, the power uprate has no impact on this analysis.
15.1.3.3	Steam System Piping Failures (Pre-trip power excursion)	Maintain coolable geometry Offsite Doses within 10CFR100 guidelines.	A combination of initial DNBR margin and reactor trip setpoints are set to minimize fuel failures. The reactor trips are credited are the Reactor Protection System (RPS) Low Steam Generator (SG) pressure trip and the CPCS VOPT. The Low SG pressure trip is not impacted by the power uprate. The CPCS VOPT trip will be adjusted to maintain the same absolute power level protection (Section 3.10.3.1). Therefore, the power uprate has no impact on any of the acceptance criteria for this event. As discussed in Section 3.12, all radiological consequences continue to meet the acceptance criteria.
15.2 Decrease in Heat Removal by the Secondary System			
15.2.1.1	Loss of External Load	Peak RCS Pressure $\leq$ 110% of Design Peak Secondary Pressure $\leq$ 110% of Design	Not analyzed since all criteria are bounded by Loss of Condenser Vacuum (UFSAR Section 15.2.1.3).

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.2.1.2	Turbine Trip	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Loss of Condenser Vacuum (UFSAR Section 15.2.1.3).
15.2.1.3	Loss of Condenser Vacuum (LOCV)	Peak RCS Pressure ≤ 110% of Design Peak Secondary pressure ≤ 110% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the High Pressurizer Pressure Trip (HPPT), which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria. As documented in the TS bases, the MSSV Inoperable Analysis (LCO 3.7.1 and LCO Table 3.7.1-1) was determined by the relationship of MSSV capacity to the sum of Rated Thermal Power and power measurement uncertainty. As that sum is unchanged, the restrictions for the various combinations of MSSVs inoperable remain unaffected.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.2.1.4	Loss of Normal AC Power	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26 and Peak LHR ≤ 21 kW/ft)	The analysis is performed at 102% power. Furthermore, the mitigating action is the CPCS Low Pump Speed Trip, which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.
15.2.2.1	Loss of External Load with a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Loss of Condenser Vacuum with Single Active Failure (UFSAR Section 15.2.2.3).
15.2.2.2	Turbine Trip with a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Loss of Condenser Vacuum with Single Active Failure (UFSAR Section 15.2.2.3).
15.2.2.3	Loss of Condenser Vacuum with a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Peak Secondary pressure ≤ 110% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the High Pressurizer Pressure Trip (HPPT), which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.2.2.4	Loss of all Normal AC Power with a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26 and Peak LHR ≤ 21 kW/ft)	Not analyzed since all criteria are bounded by reactor coolant shaft seizure (UFSAR Section 15.3.3.1)
15.2.2.5	Loss of Normal Feedwater Flow	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the Low SG Level Trip (LSGLT), which is not impacted by the power uprate. The initial SG level is the maximum SG level which is not impacted by the uprate. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.2.3.1	Feedwater System Pipe Breaks	Peak RCS Pressure ≤ 120% of Design Peak Secondary Pressure ≤ 110% of Design No Liquid release through the PSV for peak RCS pressure case	The analysis is performed at 102% power. Furthermore, the limiting case is tripped by the High Pressurizer Pressure Trip (HPPT) and the Low SG Level Trip, which are not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.
15.2.3.2	Loss of Normal Feedwater Flow with a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the Low SG Level Trip (LSGLT), which is not impacted by the power uprate. The initial SG level is the maximum SG level which is not impacted by the uprate. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.3 Decrease in Reactor Coolant Flow Rate			
15.3.1.1	Partial Loss of Forced Reactor Coolant Flow	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26 and Peak LHR ≤ 21 kW/ft)	The Partial Loss of Forced Flow was not analyzed because it is bounded by the Total Loss of Flow (UFSAR Section 15.3.2.1).
15.3.2.1	Total Loss of Forced Reactor Coolant Flow	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26 and Peak LHR ≤ 21 kW/ft)	The event involves preserving DNBR margin such that the consequences of the event do not violate the acceptance criteria. Furthermore, the mitigating action is the CPCS Low Pump Speed Trip, which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.
15.3.2.2	Partial Loss of Forced Reactor Coolant Flow with Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Maintain Coolable Geometry Peak Secondary Pressure ≤ 110% of Design	Not analyzed since all acceptance criteria are bounded by the Single Reactor Coolant Pump Sheared Shaft event (UFSAR Section 15.3.3.1).

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.3.3.1	Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design Maintain Coolable Geometry Offsite Doses a small fraction of 10CFR100 guidelines	The event involves preserving DNBR margin such that the consequences of the event do not violate the acceptance criteria. The mitigating action is provided by either the CPCS Low Pump Speed trip or the RPS Differential Pressure Low Flow Trip. Neither of these trips are impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria for this event. As discussed in Section 3.12, all radiological consequences continue to meet the acceptance criteria.
15.4 Reactivity and Power Distribution Anomalies			
15.4.1.1	Uncontrolled CEA Withdrawal at Subcritical	Peak RCS Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26 and Peak LHR ≤ 21 kW/ft)	The analysis is performed at subcritical. Depending on which CEA Banks are involved, the mitigating actions are provided by either the High Log Power Trips (Section 3.10.1) or the removal of the CPC Zero Power Bypass. The impact of power uprate was evaluated and margin in the analysis of record was sufficient to bound the change in peak heat flux, peak linear heat, and minimum DNBR. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.4.1.2	Uncontrolled CEA Withdrawal at Low Power	Peak RCS Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26 and Peak LHR ≤ 21 kW/ft)	A combination of Preserved DNBR margin and the CPCS filters are set to preclude fuel failures. The trip credited for this event is the floor of the CPCS VOPT. As discussed in Section 3.10.3.1, this trip is being modified to maintain the same absolute power level changes as the current configuration. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.
15.4.1.3	Uncontrolled CEA Withdrawal at Power	Peak RCS Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26 and Peak LHR ≤ 21 kW/ft)	A combination of Preserved DNBR margin and the CPCS filters are set to preclude fuel failures. The filter verification is impacted by the rate of change of power and not the initial power and is thus not adversely impacted by power uprate. The trip credited for this event is the VOPT. As discussed in Section 3.10.3.1, this trip is being modified to maintain the same absolute power level changes as the current configuration. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.4.1.4	Control Element Assembly Misoperation	Peak RCS Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26 and Peak LHR ≤ 21 kW/ft)	The event involves preserving DNBR margin such that the consequences of the event do not violate the acceptance criteria. The required thermal margin for the event is the ratio of the available thermal margin at the start of the event to the available thermal margin at the termination of the event. Since the choice of initial power equally affects the initial and final conditions for these events, the choice of initial power becomes insignificant. Therefore, the power uprate has no impact on any of the acceptance criteria for this event. (Note that as shown in UFSAR Table 15.4-9, this event was initiated at the nominal full power value of 3410 MWt. This representative case is still valid because the initial power is insignificant.)
15.4.1.5	CVCS Malfunction	Time after Boron Dilution Alarm for operator Action ≤ 15 minutes	This is not a Mode 1 event. Therefore, it is not impacted by the power uprate.
15.4.1.6	Startup of an Inactive Reactor Coolant System Pump	Shutdown % > 0.0	Per Technical Specifications the reactor must be subcritical if all four pumps are not operational. Therefore, this event is not impacted by the power uprate.
15.4.1.7	CEAW Mode 3, 4 and 5, All FLCEAS on Bottom	Peak RCS Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥ 1.26 and Peak LHR ≤ 21 kW/ft)	The analysis is performed at subcritical. The mitigating actions are provided by the removal of the CPC Zero Power Bypass. The impact of power uprate was evaluated and margin in the analysis of record was sufficient to bound the change in peak heat flux, peak linear heat, and minimum DNBR. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.4.3.1	Inadvertent Loading of a Fuel Assembly into an Improper Position	N/A	Most misloading events would be detected during low power startup testing. These misloading events are not impacted by power uprate. A small number of misloading events would be undetectable during startup testing and might cause an increase in core peaking as burnable poison shims burn out during power operation. The consequences of these misloads are limited by the initial DNBR margin. Therefore, these events are not adversely impacted by power uprate.
15.4.3.2	Control Element Assembly (CEA) Ejection	Peak RCS Pressure ≤ 110% of Design Centerline enthalpy of hottest fuel pellet ≤ 280 cal/gm (fuel failure threshold: total average enthalpy of hottest fuel pellet ≤ 200 Cal/gm, total centerline enthalpy of hottest fuel pellet ≤ 250 Cal/gm, DNBR < 1.26) Offsite Doses within 10CFR100 guidelines	This analysis is performed at a spectrum of initial power levels. The trip credited for this event is the VOPT. As discussed in Section 3.10.2.1, this trip is being modified to maintain the same absolute power level changes as the current configuration. Therefore, the power uprate has no impact on any of the acceptance criteria for this event. As discussed in Section 3.12, all radiological consequences continue to meet the acceptance criteria.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.5 Increase in Reactor Coolant Inventory			
15.5.1.1	Chemical and Volume Control System Malfunction	Peak RCS Pressure ≤ 110% of Design	The transient was performed at 102% power. The mitigation action was a High Pressurizer Pressure Trip (HPPT), which is not affected by power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria for this event.
15.5.1.2	Inadvertent Operation of the ECCS During Power Operation	Peak RCS Pressure ≤ 110% of Design	Not analyzed since the shutoff head of the safety injection pumps is lower than the low pressurizer pressure trip setpoint.
15.5.2.1	Chemical and Volume Control System Malfunction With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	The transient was performed at 102% power. The mitigation action was a HPPT, which is not affected by power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria for this event..
15.6 Decrease in Reactor Coolant Inventory			
15.6.3.1	Primary Sample or Instrument Line Break	Offsite Doses a small fraction of 10CFR100 guidelines	The transient was performed at 102% power. Therefore, the power uprate has no impact on any of the acceptance criteria for this event. As discussed in Section 3.12, all radiological consequences continue to meet the acceptance criteria.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.6.3.2	Steam Generator Tube Rupture	Offsite Doses a small fraction of 10CFR100 guidelines (with no iodine spike). Offsite Doses within 10CFR100 guidelines (with pre-existing iodine spike). Control Room Doses within 10CFR100 Appendix A GDC 19 guidelines.	The transient was performed at 102% power. The mitigation action was a Low Pressurizer Pressure trip which is not affected by power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria for this event. As discussed in Section 3.12, all radiological consequences continue to meet the acceptance criteria.
15.6.3.3	LOCA	10CFR50.46 Offsite Doses within 10CFR100 guidelines. Control Room Doses within 10CFR100 Appendix A GDC 19 guidelines.	No impact, see section 3.10.2 As discussed in Section 3.12, all radiological consequences continue to meet the acceptance criteria.
15.6.3.4	Inadvertent Opening of a Pressurizer Safety Valve	This event is bounded by LOCA.	This event is bounded by LOCA.
15.7 Radioactive Release from a Subsystem or Component			
15.7.3.1	Radioactive Waste Gas System Leak or Failure	Offsite Doses within 10CFR100 guidelines.	The maximum RCS and waste gas system activity is limited by the Technical Specifications. The TS limit is not changing, thus the consequence results remain bounding.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.7.3.2	Radioactive Waste System Leak or Failure (Release to Atmosphere)	Offsite Doses within 10CFR100 guidelines.	The maximum RCS activity is limited by the Technical Specifications. The TS limit is not changing, thus the consequence results remain bounding.
15.7.3.3	Postulated Radioactive Releases due to Liquid Tank Failures	Offsite release limited to 10CFR20 Appendix B.	The maximum RCS activity is limited by the Technical Specifications. The TS limit is not changing, thus the consequence results remain bounding.
15.7.3.4	Design Basis Fuel Handling Accident Inside Fuel Building	Offsite Doses within 10CFR100 guidelines. Control Room Doses within 10CFR100 Appendix A GDC 19 guidelines.	As discussed in Section 3.12, all radiological consequences continue to meet the acceptance criteria.
15.7.3.5.1	Spent Fuel Cask Drop into Spent Fuel Pool	Offsite Doses within 10CFR100 guidelines.	Since the cask handling crane is prohibited from traveling over the spent fuel pool this is not a creditable accident.

TABLE 3.10.3-1 – IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
UFSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.7.3.5.2	Spent Fuel Cask Drop to Flat Surface	Offsite Doses within 10CFR100 guidelines.	The information used for the spent fuel cast drop on a flat assessment is not impact by the Appendix K uprate. Thus, the information has not changed.
15.8 Anticipated Transient Without Scram (ATWS)			
	ATWS	Offsite Doses within 10CFR100 guidelines	The Waterford 3 ATWS mitigating systems required by 10CFR50.62 include diverse reactor trip system, diverse turbine trip, and diverse emergency feedwater actuation system. The Appendix K uprate will not affect these diverse mitigating systems.
15.9 Miscellaneous			
15.9.1.1	Asymmetric Steam Generator Transient	No Fuel Failure (Minimum DNBR $\geq$ 1.26 and Peak LHR $\leq$ 21 kW/ft)	The event involves preserving DNBR margin such that the consequences of the event do not violate the acceptance criteria. The mitigation action was a CPCS Auxiliary trip (e.g. CPCS ΔT trip) which is not impacted by power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.

3.10.4 Steam Generator Water Level

The small change in nominal steam pressure and feedwater temperature due to the power uprate conditions does not change the final calculated steam generator water level channel uncertainties. Other potential contributors to level measurement uncertainty, including recirculation ratio, reference leg temperature effects, were found to be not significantly affected by the proposed uprating. Therefore, the uprate does not necessitate changes to the uncertainties for the steam generator water level trip(s). Refer to Section 3.10.1 for additional detail on steam generator level setpoints.

3.11 CONTAINMENT/BOP ACCIDENT EVALUATIONS

3.11.1 Mass and Energy Release Data

3.11.1.1 LOCA Mass and Energy Release Data for Subcompartment Pressurization

Containment analyses demonstrate the adequacy of the Containment Building and its internal walls, and qualify the equipment inside containment for a design basis accident. A LOCA analysis was evaluated to determine compartment pressurization of subcompartments located inside containment. This section discusses the impact of the 1.5 percent uprate on the subcompartment LOCA mass and energy analyses.

The subcompartment mass and energy release data was generated by the NRC approved code CEFLASH-4A. A LOCA analysis that supplies mass and energy for subcompartment pressurization was evaluated to determine compartment pressurization of subcompartments located inside containment. The initial RCS temperature values are the primary parameters that could effect the subcompartment mass and energy release rates. This is because the peak differential pressure will occur before reactor power or secondary energy can have much effect on the stored energy in the RCS. (Note that the run times for the cases are only 4 seconds). The use of lower temperatures will maximize the mass out a given break. The Appendix K uprate maintains the same cold leg temperature. The current mass and energy release data remains applicable for the 1.5 percent uprate for the Subcompartment Pressurization analyses.

3.11.1.2 LOCA Mass and Energy Release Data Containment Response

The current mass and energy release data for input into the containment response analysis were generated at a power level of 3,734 MWt. This data bounds both the current and uprate power levels. The 3,734 value is based on 108 percent of the current licensed power level plus an additional 2 percent that accounts for measurement inaccuracy.

Critical parameters related to the mass and energy release to containment during a LOCA are provided in the following table. Table 3.11.1.2-1 shows that existing analysis initial conditions bound all but the steam generator pressure and liquid mass assumptions for a 1.5 percent power uprate. These two parameters affect the stored energy in the steam generator at the initiation of the reflood phase. A simplified energy calculation demonstrated that the difference in steam generator energy at the initiation of the reflood phase is just above 1.3 percent. This is considered insignificant for this analysis. The results of the current containment LOCA mass and energy release data used for input to the containment response analysis at 3,734 MWt remain applicable for the 1.5 percent power uprate.

3.11.1.3 Steam Line Break Mass and Energy Releases Inside and Outside Containment

The 1.5 percent power uprate has the potential to effect 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 steam line break was performed from a core power of 3,457.8 MWt (see Table 3.11.1.3-1). The results remain applicable for the 1.5 percent power uprate.

Critical parameters related to the mass and energy release during a steam line break event is provided in the following table. Table 3.11.1.3-1 shows that existing analysis initial condition assumptions bound as much as a 1.7 percent power uprate.

Table 3.11.1.2-1
Containment LOCA Mass & Energy Analysis Assumptions vs. Operating Point Values

PARAMETER	Appendix K Uprate Operating Point	LOCA M & E input assumed value	Is analysis bounding ?
Core Power MWt	3,448.	3734.4	YES
Primary Bulk T_h, °F	600.2	614.9	YES
Primary T_c, °F	545.0	552.0	YES
Primary Flow Rate, lbm/sec	44,522.4	41,274	YES
Primary Pressure, psia	2250	2310.0	YES
Feedwater Temperature, °F	442.7	445.8	YES
Feedwater Enthalpy, BTU/lbm	422.2	425.7	YES
Steam Pressure, psia	831.5	817	Insignificant**
SG Liquid Mass, lbm	179,983*	178,490	Insignificant**

* Estimated SG liquid mass at time of reflood.
100% steam flow until Turbine Stop Valves close at 1.25 sec.
100% feed flow until main feedwater isolation at 11.0 sec.
100% feed flow / steam flow mismatch for 11 – 1.25 = 9.75 sec
Appendix K Uprate initial SG liquid mass = 159,158 lb.
Appendix K Uprate initial feedwater flow rate = 2,135.9 lb./sec.
159,158 + (9.75*2,135.9) = 179,983 lb.

** Steam generator energy at the initiation of the reflood phase
Enthalpy at 831.5 psia = 515.4 Btu/lb. Energy = 179983 * 515.4 = 92763251 Btu
Enthalpy at 817.0 psia = 512.9 Btu/lb. Energy = 178490 * 512.9 = 91547521 btu

The difference in SG sensible energy at the initiation of the reflood phase is approximately 1.3 percent higher for the proposed 1.5 percent power uprate. However, the impact of slightly higher SG energy on the mass and energy releases and, consequently, containment response is deemed to be negligible.

Table 3.11.1.3-1
Containment SLB Mass & Energy Analysis Assumptions vs. Operating Point Values

PARAMETER	Appendix K Uprate Operating Point	MSLB M & E analysis input assumed value	Is analysis bounding ?
Core Power MWt	3,448.	3457.8 (1)	YES
Primary T_c, °F	545.0	560	YES
Primary Flow Rate, lbm/sec	44,522.4	48,946.1 (2)	YES
Primary Pressure, psia	2250	2310.6	YES
Feedwater Enthalpy, BTU/lbm	422.2	427.4	YES
Steam Pressure, psia	831.5	859	YES
SG Total Mass**, lbm	174,030.	182,521.8 (3)	YES
SG Liquid Mass (lbm)	159,158.	166,086.8 (4)	YES

** Includes mass in steam lines from SG to MSIV (approximately 2500 lbm)
(1) Core Power = 3,390*1.02 = 3457.8
(2) 475200 gpm *46.23lb/ft^3 / (60 sec. * 7.4805g/ft^3) = 48,946.14 lb/sec
(3) SG Total Mass = (332173.7 + 32869.95)/2 = 182,521.825
(4) SG Liquid Mass = 332173.7/2 = 166,086.85

3.11.2 Containment Analysis

3.11.2.1 MSLB and LOCA

As stated in Section 3.11.1.2, the mass and energy release data for the LOCA bound the power uprate conditions. Therefore, the peak LOCA containment pressure and temperature will not be impacted by the power uprate. The containment heat removal systems capability to reduce the containment pressure by one half of the peak within 24 hours following a LOCA is also not impacted by the power uprate.

As stated in Section 3.11.1.3, the mass and energy release data for the Steam Line Break bound the power uprate conditions. Therefore, the peak Steam Line Break containment pressure and temperature will not be impacted by the power uprate.

3.11.3 Equipment Qualification Accident Environments

As stated in Section 3.11.2, the current containment LOCA and main steam line break analyses will not be affected by uprate conditions. The current equipment qualification accident environments inside containment bound the environments resulting from the power uprate.

3.11.3.1 LOCA and Main Steam Line Break Inside Containment

As stated in Section 3.11.2, the current containment LOCA and main steam line break analyses will not be affected by uprate conditions. The current equipment qualification accident environments inside containment bound the environments resulting from the power uprate.

3.11.3.2 High-Energy Line Breaks Outside Containment

The Waterford 3 Design and Licensing Basis does not include any High Energy Line Breaks Outside of Containment. See Section 3.8.10 above for a discussion of Balance Of Plant Piping.

3.12 RADIOLOGICAL CONSEQUENCES

The current licensed core power level for Waterford 3 is 3,390 MWt. The post-accident radiological analyses were originally based upon at least 1.02 times the licensed core level.

The radiological source terms considered relate to non-LOCA design basis accidents, fuel handling accident, and that resulting from the Maximum Hypothetical Accident (MHA). (The MHA source terms are also used in LOCA and EQ evaluations.)

For the fuel handling accident and MHA source terms, the radiological analyses currently supporting normal operation are based on a core power level of 102 % or higher with an 18-month operating cycle. Therefore, the 1.5 percent uprate conditions are covered by the existing analyses.

For evaluation of radiological consequences following non-LOCA design basis event, the radiological source terms were divided into fuel failure and non-fuel failure events. The radiological source terms for the fuel failure events were based upon a maximum radial peaking factor and a core power of 3,390 MWt. However since current design constraints limit the hot rod radial power peaking factor to lower than the assumed maximum, the current non-LOCA source term will be applicable up to 1.5 percent uprate conditions.

The non-fuel failure non-LOCA transient source terms were based upon or exceeded the allowable Technical Specification RCS activity limits. The NSSS steam activity release rates were based upon at least 1.02 times the licensed core power. In addition the allowable Technical Specification RCS activity limits will remain at their current value. Thus, the non-fuel failure transients remain bounded by existing analyses.

3.13 NUCLEAR FUEL

This section summarizes the evaluations performed to determine the effect of the 1.5 percent uprating on the nuclear fuel. The core design for Waterford 3 is performed for each specific fuel cycle and varies according to the needs and specifications for each cycle. However, some fuel-related analyses are not cycle specific. The nuclear fuel review for the 1.5 percent uprate, 3,441 MWt, evaluated the fuel core design, thermal-hydraulic design, and fuel rod performance.

3.13.1 Fuel Core Design

The effects of the 1.5 percent uprate conditions 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. Since the power uprate is relatively small, the representative cycles are adequate to demonstrate the sensitivity of reload parameters to the power uprate conditions. The expected ranges of variation in key parameters were determined. The methods and core models used in the uprate analyses are consistent with those presented in the Waterford 3 UFSAR. No changes to the nuclear design philosophy, methods, or models are necessary due to the uprating. The core analyses for the uprating were performed primarily to determine if the values previously used for the key safety parameters remain applicable prior to the cycle-specific reload design.

The core analyses show that the implementation of the power uprate does not result in changes to the current nuclear design basis documented in the UFSAR. The impact of the uprate on peaking factors, rod worths, reactivity coefficients, shutdown margin, and kinetics parameters is either well within normal cycle-to-cycle variation of these values or controlled by the core design and will be addressed on a cycle-specific basis consistent with reload methodology.

3.13.2 Core Thermal-Hydraulic Design

The core thermal-hydraulic design and methodology were evaluated at the uprated core power level of 3,441 MWt. The thermal hydraulic design is based on the TORC computer code described in Reference 3.13.2-1, the CE-1 Critical Heat Flux (CHF) correlation described in References 3.13.2-2 and 3.13.2-11, the simplified TORC modeling methods described in Reference 3.13.2-3, and the CETOP code described in Reference 3.13.2-4. In addition, the DNBR analysis uses the methodology for determining the limiting fuel assembly(ies).

The Modified Statistical Combination of Uncertainties (MSCU) presented in Reference 3.13.2-5 was applied to validate the design limit of 1.26 on the CE-1 minimum DNBR. This DNBR limit includes the following allowances:

1. NRC specified allowances for TORC code uncertainty and the CE-1 CHF cross correlation validation uncertainty as discussed in Reference 3.13.2-10.
2. NRC imposed 0.01 DNBR penalty for HID-1 grids as discussed in References 3.13.2-6 through 3.13.2-8.
3. Rod bow penalty equivalent to 1.75% on minimum DNBR as discussed in Reference 3.13.2-9.

The core thermal-hydraulic design and methodology remain applicable at the uprated core power level of 3,441 MWt.

3.13.3 Fuel Rod Design

The thermal performance of erbia and UO_2 fuel rods for a 1.5 percent power uprated Waterford 3 core were evaluated using the FATES3B version of the CENP fuel evaluation model, the erbia burnable absorber methodology described in Reference 3.13.3-4, and the maximum pressure methodology described in Reference 3.13.3-5. This 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 maximum predicted fuel rod internal pressure for the uprated core remains below the critical pressure for No-Clad-Lift-Off (Reference 3.13.3-5).

4.0 MISCELLANEOUS

4.1 AFFECTED PLANT PROGRAMS

The power uprate has the potential to affect programs that are developed and implemented by station personnel to demonstrate that topical areas comply with various design and licensing requirements. The plant programs and/or issues listed in Table 4.1-1 were reviewed to determine the impact due to the power uprate. In addition to the programs, plant Technical Specifications address specific requirements for a number of these programs. The programs that have Technical Specifications associated with them are identified in Table 4.1-2.

For the programs listed in Table 4-1, the controlling procedures and processes for the programs and key reference items within the procedures were reviewed. Program sponsors, implementing organization personnel and other cognizant individuals were interviewed for those issues and programs that could potentially be impacted by the uprate. Based upon the review of this information, the extent of impact by the implementation of the power uprate was determined for the various issues and programs.

For the programs listed in Table 4.1-2, the Technical Specifications and Technical Requirements Manual Sections associated with the programs were reviewed to identify any areas affected by power uprate.

The review results are summarized in the tables using two groupings: not affected; and requires update. The review identified two programs that would be impacted by the uprate. Changes to these programs will be captured by in place change processes as identified below:

4.1.1 Simulator

The Waterford 3 simulator mimics the actual control room and is primarily used for training of operations personnel. In addition to the overall physical likeness between the actual control room and the simulator, computer systems provide simulator responses that are intended to match actual plant conditions for the simulation of accidents and transients, to the greatest extent possible. To ensure that the simulator accurately reflects the plant status, physical appearance (hardware) and simulation of plant response (software), changes resulting from the power uprate must be effectively communicated.

A review of the training simulator fidelity with the new power rating will be included at the next regularly scheduled review following the uprating in RTP. Simulator revalidation is performed in accordance with ANSI/ANS 3.5-1985.

Physical changes (hardware) that affect the control room and the simulator will be implemented through plant approved change processes. Copies of these change processes are procedurally routed to the Training Department and the simulator personnel implement appropriate changes.

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.

The implementation of the power uprate will also result in changes in plant operating characteristics (software changes). These changes will range from simple changes in process temperatures and flow rates to plant responses to accidents and transients.

4.1.2 Flow Accelerated Corrosion (FAC)

The main feedwater systems, as well as other power conversion systems, are important to safe operation. Failures of passive components in these systems, such as piping can result in undesirable challenges to plant safety systems required for safe shutdown and accident mitigation. Failure of high-energy piping, such as feedwater system piping, can result in complex challenges to operating staff and the plant because of potential system-interactions of high-energy steam and water with other systems, such as electrical distribution, fire protection, and security. Waterford 3 has committed to adhere to criteria, codes and standards for high-energy piping systems described in licensing documents. Such commitments are a part of the licensing basis for the facility. An important part of this commitment is that piping will be maintained within allowable thickness values.

FAC, in the piping systems at Waterford 3, is modeled using the CHECWORKS computer program. CHECWORKS models will be revised, as appropriate, to incorporate flow and thermodynamic states that are projected for uprated conditions. The results of these models will be factored into future inspection/pipe replacement plans consistent with the current FAC Program requirements.

4.2 OPERATING PROCEDURES (ABNORMAL/NORMAL) AND OPERATOR ACTIONS

4.2.1 Control Room

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 the 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% 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.

4.2.2 Normal Operating Procedures/Emergency Operating Procedures/Off-Normal Procedures

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% value for rated thermal power, 100% Licensed Power Limits, Reactor Coolant system delta temperature, 100% 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 training below. The Waterford 3 Technical Requirements Manual will be revised for the LEFM CheckPlus out of service power reduction described in Section 3.2.

4.2.3 Operator Training and Simulator

Classroom and Simulator training will be provided on all changes that affect operator performance caused by this power uprate. Changes to the simulator will be made consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be validated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." All Control Room and plant process computer system changes a s a result of the power uprate will be completed.

4.3 STATION BLACKOUT EVENT

On April 14, 1989, Entergy submitted the response for Waterford 3 to the station blackout rule 10CFR 50.63. The response was prepared based on the calculation developed using the guideline outlined in NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors."

The general criteria states that procedures and equipment relied upon in a station blackout event should ensure that satisfactory performance of necessary decay heat removal systems is maintained for the required 4 hour coping duration. The core must remain covered and containment integrity should be provided to the extent that isolation valves perform their intended functions without AC power.

Although there is a slight increase in decay heat generation (slightly higher cooling load during cooldown) for the proposed uprate from 100% to 101.5% (3,390 to 3,441 MWt) 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% power (3,390 MWt) 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 resulted 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 Technical Specification minimum requirement of 170,000 gallons for the condensate storage pool, thus the plant's current condensate inventory is adequate.

The Atmospheric Dump Valves 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 % RTP +2 % instrument uncertainty. This design bounds the power uprate.

Other elements of the SBO analysis have not significantly changed: Plant Lighting, RCS Inventory Loss, Shutdown Margin, Containment Isolation, Loss of Ventilation, Compressed Air, Battery Capacity, Coping Period, Diesel Generator Reliability, or equipment required operable for Station Blackout. None of the SBO associated instruments require control setpoint changes, and none of the associated instruments exceed design basis due to the power uprate. Therefore, the SBO analysis is not affected by this power uprate.

4.4 SAFETY RELATED VALVES

4.4.1 Generic Letter 89-10 "Safety Related Motor-Operated Valve Testing and Surveillance"

There are no required changes to the Waterford 3 GL 89-10 MOV Program as a result of the 1.5-percent power uprate. The applicable Design Basis calculations have been reviewed. The Design Basis upstream, downstream, differential pressures and flow used for MOV sizing were developed from conservative assumptions, which either were not affected by or are bound by the uprate conditions. No changes to the margin of safety inherent in these calculations have been made. In addition, no setpoint changes will be required as a result of the power uprate.

4.4.2 Generic Letter 95-07 "Pressure Locking and Thermal Binding of Safety Related Operated Gate Valves"

A review of the documentation and evaluations of GL 95-07 was performed to determine if the proposed 1.5-percent power increase would adversely affect any conclusions or qualifications that were approved by the NRC upon closure of the subject Generic Letter.

The Design Basis conditions used to determine susceptibility to Pressure Locking were developed from conservative assumptions, which either were not affected by or are bound by the uprate conditions. The conditions detailed in the evaluation remain bounding for the 1.5 percent power uprate. Conditions, conclusions and the bases for these conclusions as originally understood by the NRC, are unchanged and remain valid.

4.4.3 Generic Letter 96-06 "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

A review of the existing documentation and evaluations of GL 96-06 was performed to determine if the proposed 1.5-percent power increase would adversely affect any of the previous conclusions related to containment integrity (i.e., relative to overpressurization of safety related, water filled, isolable piping sections inside containment) and water hammer in the Containment Fan Coolers or their supply and return piping.

Conditions detailed in the evaluation remain bounding for the 1.5-percent power uprate. The post accident environments inside containment have-not changed as a result of the power uprate and there are no physical changes to the Containment Fan Coolers and / or associated piping. Therefore the isolable piping sections are not impacted.

4.4.4 Air Operated Valves

Waterford 3 completed the Air Operated Valve (AOV) Program valve scoping based on Probabilistic Safety Analysis (PSA) ranking and utilizing an AOV Expert Panel. There are three categories in the scope. Category (CAT) 1 AOVs are safety or non-safety related valves that have High or Medium PSA ranking. CAT 2 AOVs are active safety-related valves with Low PSA ranking and non-safety valves that are trip critical. A review concluded that the 1.5 percent Appendix K Power Uprate has no affect on CAT 1 & 2 AOVs.

4.5 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

In compliance with 10CFR50.62, ATWS mitigation system actuation circuitry has been incorporated into the design of Waterford 3. The ATWS mitigation system has been reviewed with respect to the proposed 1.5 percent power uprate and no changes are needed (see Table 3.10.3-1 Section 15.8 "Anticipated Transients Without Scram").

4.6 RESPONSE TO PREVIOUS NRC UPRATE RAI ON INDEPENDENT PLANT EVALUATION

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.

4.7 FIRE PROTECTION

The Plant Fire Protection System (FP) provides fire protection for the systems and equipment throughout the plant. The Plant Fire Protection System is designed to provide:

- A reliable supply of water of suitable quality for fire fighting purposes, in quantities sufficient to satisfy the maximum probable demand.
- A reliable pumping system for delivering this water to all hose stations and sprinklers at the required flow rates and residual pressures.
- A sufficient number of yard fire hydrants strategically located to provide large hose stream protection for all station buildings and other fire hazards in the yard area.
- Standpipe connections for fire hose streams located in areas throughout the station.
- Hand type portable fire extinguishers of the proper types located throughout all areas of the station to provide a first defense against small incipient fires.
- Automatic or manual sprinkler systems installed where warranted based on fire hazard analysis.

The fire suppression system consists of two storage tanks, three fire water pumps, a jockey pump and the associated piping and valves to provide the capability of supplying water to any sprinkler, standpipe, or hydrant.

The jockey pump maintains system pressure. The other three pumps provide water for the fire suppression system. The fire pump section piping and valving arrangement is designed so that all fire pumps can take suction from either or both tanks. Also a leak in one tank will not cause the other to drain. The firewater pumps discharge into a water distribution system.

The water distribution system consists of underground yard piping serving all plant yard fire hydrants, sprinkler systems, water spray systems and interior standpipe systems. The underground piping forms a complete fire loop around the plant. Post indicator type sectionalizing control valves are installed in the main fire loop to facilitate system maintenance and repair without placing the entire loop out of service. Branch connections from the fire main to all systems are provided with isolation valves to minimize the need for closing sectionalizing valves on the main fire loop.

Yard fire hydrants are connected to the fire main loop at intervals of approximately 250 ft. The main fire loop supplies two other fire loops; the reactor auxiliary building, which in turn supplies the containment building, and the turbine building. The Fuel Handling Building, Service Building, Administrative Building, and Maintenance Support Building are supplied by individual taps.

The combustible equipment and new or existing penetrations etc. that are being installed/modified to support the 1.5 percent power uprate have been evaluated, with respect to impact on plant fire protection. The results of the evaluation are that the 1.5 percent Appendix K Power Uprate has no affect on plant fire protection.

4.8 RADIOACTIVE WASTE SYSTEMS

Radioactive wastes are processed through either the Solid Waste Management System the Liquid Waste Management System or the Gaseous Waste Management System. The original design of these systems was based on reactor coolant radioisotope concentrations using one percent failed fuel and a RTP of 3,560 MWt. The proposed power uprate is to a RTP of 3,441 MWt and the fuel design and maximum burnup and thus the probability of fuel failure will be unaffected by the uprate. The original design of these systems thus bounds the proposed 1.5 percent power uprate.

4.9 RADIATION PROTECTION

The original Waterford 3 Radiation Protection design was based on a Rated Thermal Power of 3,560 MWt and one percent failed fuel. As stated above, in Section 4.9, the proposed power uprate will not change the probability of fuel failure. The original Radiation Protection Design Basis thus bounds the proposed 1.5 percent power uprate to a RTP of 3,441 MWt. In addition the Fuel Handling Building was reanalyzed for the 1998 Waterford 3 Spent Fuel Pool Rerack Project (See Section 3.5.11). This reanalysis assumed a RTP of 3,661 MWt. This analysis also bounds the proposed power uprate to 3,441 MWt.

4.10 HEATING, VENTILATION AND AIR CONDITIONING SYSTEMS

4.10.1 Control Room Heating, Ventilation and Air Conditioning System

The purpose of the Control Room Heating, Ventilation, and Air Conditioning (HVC) system is to maintain the Control Room envelope in a habitable condition. This envelope includes the Control Room, computer room, computer room supplementary air conditioning equipment room, HVC equipment room, emergency living quarters, emergency storage room, toilets, locker rooms, kitchen, kitchenette, supervisor's office, corridors, conference room, and vault. Control Room Habitability systems are required to assure that the operators can remain in the Control Room to operate the plant safely under normal conditions and maintain the unit in a safe condition under accident conditions.

Waterford-3 is geographically located in an industrial area and is where the potential exists for fires, explosions or releases of toxic gases due to the transported and stored volumes of chemicals in the immediate vicinity. The HVC system is designed to establish and maintain a habitable atmosphere in the event of an FSAR analyzed toxic chemical accident or a design base accident (DBA) with its resulting radioactive environment.

The system consists of two full capacity redundant Air Handling Units (AHUs), two full capacity toilet exhaust fans, a kitchen and conference room exhaust fan, two full capacity redundant Control Room Emergency Filtration Units, redundant isolation valves for the two emergency intakes, the normal intake and the two separate exhausts. Other HVC equipment includes individual area dampers, heaters and redundant equipment room AHUs.

The HVC system boundary is defined as the Control Room Envelope. This envelope is controlled by automatically initiated isolation features through Safety Injection Actuation Signals (SIAS), High Radiation Signals, and Toxic Gas Signals. The system maintains a slight positive pressure, relative to the outside atmosphere, within the envelope during normal operations to prevent any outside air from bypassing the safety related monitoring instrumentation located in the air intake path. A variety of purging operations may be conducted through various dampers and exhaust flow paths. These purging operations may be performed to remove smoke. The North and South emergency air intakes provide two sources of Control Room fresh air during radiological accident conditions.

The proposed 1.5 percent power uprate does not change the probability or severity of an offsite chemical release. The original design bases for the onsite radiological releases were based on a RTP of at least 102 % and thus bound the proposed 1.5 percent power uprate. The proposed 1.5 percent uprate therefore has no affect on the Control Room Heating, Ventilation and Air Conditioning System.

4.10.2 Reactor Auxiliary Building Ventilation 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 RAB Normal Ventilation System is designed to meet the following requirements:

- Maintain a suitable operating environment for all equipment and personnel during normal operation.
- Maintain the air flow from areas of low potential radioactivity to areas of progressively higher potential radioactivity.
- Limit the concentration of airborne radioactivity by circulating a sufficient volume of purging air.
- Minimize airborne fission product releases from the building exhaust during normal operations.
- Monitor ventilation system discharge to detect and prevent the excessive release of airborne radioactivity.
- Provide a means for filtering containment purge air.
- Permit periodic inspections and testing of the system's components.

The heat loading in the RAB is not affected by the proposed 1.5 percent power uprate because the base loads (motor losses, lighting etc.) remain unchanged and the heat loads due to accident conditions were originally based on a Rated Thermal Power of 102 % or greater. The proposed 1.5 percent power uprate therefore has no impact on the RAB Ventilation System.

Table 4.1-1 Program Issues

Issues and Programs	Requires Update
Plant Simulator	YES
Fire Protection (Appendix R)	NO
Check Valves	NO
Motor-Operated Valve Administrative Program (GL 89-10)	NO
Air-Operated Valves	NO
River/Service Water System Control and Monitoring (GL 89-13)	NO
Inservice Inspection Program	NO
Inservice Test Program	NO
Equipment Qualification	NO
Human Factors	NO
Station Blackout	NO
Anticipated Transient Without Scram	NO
Flow-Accelerated Corrosion Program	YES

Table 4.1-2 Technical Specification Programs

Program	Requires Update
Secondary Water Chemistry Program (TS 6.8.5a)	NO
Radioactive Effluent Controls Program (TS 6.8.6a)	NO
Radioactive Environmental Monitoring Program (TS 6.8.6b)	NO
Radiation Protection Program (TS 6.11)	NO
Process Control Program (TS 6.13)	NO
Containment Leakage Rate Testing Program (TS 6.17)	NO

No - Programs not impacted by uprate change or are bounded by existing-analysis.
Yes - Programs impacted and changes to be addressed in uprate implementation.

5.0 ENVIRONMENTAL IMPACT EVALUATION

Protection of the environment is assured by compliance with permits issued by federal, state, and local agencies.

The environmental review conducted for the proposed power uprate assessed the existing operating license and Louisiana Pollutant Discharge Elimination System (LPDES) permit limits and the information contained in the Final Environmental Report (FER). This assessment included determining whether the power uprate would cause the plant to exceed discharge limitations or LPDES permit conditions associated with the operation of the plant. In addition, a review of the recent Waterford 3 Annual Radioactive Effluent Discharge Reports demonstrates that the actual releases from the plants are a very small percentage of the Technical Specification allowable limits and the FER estimates. The discharge amounts will not be significantly increased by the thermal power uprate and will continue to be a small percentage of the allowable limits and the FER estimates.

Onsite and offsite radiation exposures from normal operation and postulated accidents are addressed in Section 3.12. The offsite doses postulated under accident conditions remain within the guidelines of 10CFR100.

The FER assessed the non-radiological impacts of plant operation as a function of plant design features, relative loss of renewable resources, and relative loss or degradation of available habitat. Environmental impacts associated with 40-year operating licenses were originally evaluated in the FER. After weighing the environmental, economic, technical, and other benefits against environmental costs and considering available alternatives, and subject to certain conditions, from the standpoint of environmental effects, the FER concluded that the issuance of the operating license for Waterford 3 was an acceptable action. These assessments, and the assumptions on which they were based, remain valid and are not impacted as a result of the thermal power uprate.

5.1 LOUISIANA POLLUTANT DISCHARGE ELIMINATION SYSTEM PERMIT IMPACT

Waterford 3 employs both closed-loop and open loop cooling systems. The closed loop cooling systems consist of the wet and dry cooling towers of the CCW and ACCW systems. These systems dissipate heat to the atmosphere. The open loop cooling system is the Circulating Water system which dissipates heat to the Mississippi River. All water used within the closed-loop cooling system is recycled except for system makeup that comes from the Water Treatment System supplied from the Parish water supply. The dry cooling towers, wet cooling tower basins and circulating water system are addressed in Sections 3.7 and 3.8.

The Waterford 3 LPDES permit (Permit No. LA0007374) places the following limits on plant discharges:

1. Flow – 1,518 Million gallons per day maximum
2. Discharge Temperature – 118 °F maximum
3. Heat Rejection – 9,500 10^6 Btu/hr maximum

The heat duty increase associated with uprate is mainly associated with the circulating water system and will be approximately 117 x 10^6 Btu/hr. This represents a 1.5 percent increase over the present heat duty, but is insignificant when compared to the current heat load from the plant. The circulating water temperature increase expected as a result of uprate will be less than 0.5 °F over existing plant operation. Therefore, the proposed thermal power uprate of Waterford 3 will have no adverse impacts on the environment and will not result in exceeding LPDES permit limits.

5.2 ENVIRONMENTAL IMPACT CONSIDERATION SUMMARY

The proposed change does not involve a significant hazards consideration, a significant change in the types of, or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c) (9). Therefore, pursuant to 10CFR51.22(b), an environmental assessment of the proposed change is not required.

6.0 REFERENCES

References for Section 1.0:

1.0-1 Caldon Engineering Report (ER) - 80P, "Topical Report - Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System", Revision 0, March 1997, Serial No. 388P.

1.0-2 Caldon Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With LEFM√™ or LEFM CheckPlus™ System", Revision 3, February 2001, Serial No. 186P.

Reference for Section 2.0:

2.0-1 Code Of Federal Regulations, Title 10, Part 50, Appendix K, "ECCS Evaluation Models".

References for Section 3.5.11:

3.5.11-1 Amendment No. 144 to Facility Operating License NPF-38 – Waterford Steam Electric Station, Unit 3, July 10, 1998.

References for Section 3.6:

3.6-1 Report CEN-367-A, "Leak-Before-Break Evaluation of Primary Coolant Loop Piping in CE Designed NSSSs," February 1991.

3.6-2 "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 120 to Facility Operating License No. NPF-38, Entergy Operations, Inc., Waterford Steam Electric Station, Unit 3, Docket No. 50-382," June 24, 1996.

References for Section 3.7:

3.7.3-1 Amendment No. 167 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3, "Addition of Main Feedwater Isolation Valves to Technical Specifications and Request for NRC Staff Review of an Unreviewed Safety Question (TAC No. MA6173)", September 5, 2000.

References for Section 3.8:

3.8.1-1 Waterford 3 Drawing No. LOU-1564 G-072 Sh. 1, "Heat Balance".

References for Section 3.10.2:

3.10.2-1 CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

CENPD-132P, Supplement 1, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," February 1975.

CENPD-132, Supplement 2-P, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1975.

CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985.

3.10.2 -2 CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," August 1974.

CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977.

CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

3.10.2-3 CENPD-254-P-A, "Post-LOCA Long Term Cooling Evaluation Model," June 1980.

References for Section 3.10.3:

3.10.3-1 NRC Regulatory Guide 1.70, Revision 2, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," September 1975.

3.10.3 -2 USNRC, "Technical Specifications Waterford Steam Electric Station, Unit No. 3, Docket No. 50-382, Appendix "A" to License No. NPF-38," Amendment No. 170.

References for Section 3.13.2:

3.13.2-1 CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April, 1986.

3.13.2-2 CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," September, 1976.

3.13.2-3 CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods," June, 1981.

3.13.2-4 CEN-160(S)-P, Rev. 1-P, "CETOP Code Structure and Modeling Methods for San Onofre Nuclear Generating Station Units 2 and 3," September, 1981.

3.13.2-5 CEN-356(S)-P-A, Rev. 01-P-A, "Modified Statistical Combination of Uncertainties," May, 1988.

3.13.2-6 CEN-155(S)-P, "CE-1 Applicability to San Onofre Units 2 and 3 HID-2 Grids, Response to NRC Questions," March, 1981.

3.13.2-7 CEN-165(S)-P, "Responses to NRC Concerns on Applicability of the CE-1 Correlation to the SONGS Fuel Design," May 1981.

3.13.2-8 NUREG-0787, Supplement 1, "Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit No. 3," Docket No. 50-382, October 1981.

3.13.2-9 CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983.

3.13.2-10 Robert A. Clark (NRC) to William Cavanaugh III (AP&L), "Operation of ANO-2 During Cycle 2," July 21, 1981 (Safety Evaluation Report and License Amendment No. 26 for ANO-2).

3.13.2-11 CENPD-207-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non-uniform Axial Power Distribution," December, 1984.

References for Section 3.13.3:

3.13.3-1 CEN-161(B)-P Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992.

3.13.3-2 CENPD-139-P-A, "Fuel Evaluation Model," July 1974.

3.13.3-3 CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.

3.13.3-4 CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

3.13.3-5 CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

ATTACHMENT 3

TO

W3F1-2001-0091

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS

IN THE MATTER OF AMENDING

LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

Docket No. 50-382

or indirectly any control over (i) the facility, (ii) power or energy produced by the facility, or (iii) the licensee of the facility. Further, any rights acquired under this authorization may be exercised only in compliance with and subject to the requirements and restrictions of this operating license, the Atomic Energy Act of 1954, as amended, and the NRC's regulations. For purposes of this condition, the limitations of 10 CFR 50.81, as now in effect and as they may be subsequently amended, are fully applicable to the equity investors and any successors in interest to the equity investors, as long as the license for the facility remains in effect.

(b) Entergy Louisiana, Inc. (or its designee) to notify the NRC in writing prior to any change in (i) the terms or conditions of any lease agreements executed as part of the above authorized financial transactions, (ii) any facility operating agreement involving a licensee that is in effect now or will be in effect in the future, or (iii) the existing property insurance coverages for the facility, that would materially alter the representations and conditions, set forth in the staff's Safety Evaluation enclosed to the NRC letter dated September 18, 1989. In addition, Entergy Louisiana, Inc. or its designee is required to notify the NRC of any action by equity investors or successors in interest to Entergy Louisiana, Inc. that may have an effect on the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level 3441

EOI is authorized to operate the facility at reactor core power levels not in excess of 3390 megawatts thermal (100% power) in accordance with the conditions specified herein.

2. Technical Specifications and Environmental Protection Plan

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The Technical Specifications contained in Appendix A, as revised through Amendment No. 171 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.

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3390 MWt.
3441

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.26 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHIELD BUILDING INTEGRITY

1.27 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed,

b. The shield building filtration system is in compliance with the requirements of Specification 3.6.6.1, and

c. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

ATTACHMENT 4

TO

W3F1-2001-0091

COMMITMENT SUMMARY

IN THE MATTER OF AMENDING

LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

Commitment Summary

COMMITMENT	TYPE*		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
New procedures for maintenance and calibration of the LEFM CheckPlus system will be developed per the design control process based on the vendor's recommendations. p1-2	X		implementation date
If the LEFM CheckPlus system is not operable the Power Limit will be administratively controlled at a level consistent with the accuracy of the available instrumentation as described in Section 3.2 below. The limiting conditions discussed above will be contained in the TRM. p1-2 & 3-6 & 3-7	X		implementation date
To further ensure this reduced power measurement uncertainty is validated and maintained, the following additional actions will be performed: • The implementing modification package specifies the affected maintenance and operating procedures that must be in place prior to declaring these units operable and raising plant power above 3,390 Mwt. • the system's software has been developed and will be maintained under a verification and validation (V&V) program.	X		implementation date

The uncertainties of the venturi and temperature element based inputs are expected to increase over time due to drift and ambient temperature uncertainty effects. These effects will be addressed through administrative controls. p3-4	X		implementation date
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 CheckPlus system. p3-6 Plant maintenance and calibration procedures will be revised to incorporate Caldon's maintenance and calibration requirements prior to declaring the LEFM CheckPlus system OPERABLE and raising power above 3,390 MWt.	X		implementation date
An Alden data report for these tests and a Caldon engineering report evaluating the test data will be on file. The calibration factor used for the LEFM CheckPlus at Waterford 3 will be based on these reports. The uncertainty in the calibration factor for the spools will be based on the Caldon engineering report. 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. p3-7	X		implementation date

The increase in primary to secondary pressure differential will be accounted for and adjusted in the SG Degradation Assessment specific to in-situ pressure testing screening criteria. The [SG] inspection program will include consideration of the higher temperatures in crack growth rate analyses. p3-29	X		implementation date
The reductions in fluence from fuel management will be measured as part of the next surveillance capsule evaluation at the end-of-cycle 11 and assessed for power uprate conditions to project reactor vessel fluence in future cycles. p3-32	X		April, 2003
CHECWORKS models will be revised, as appropriate, to incorporate flow and thermodynamic states that are projected for uprated conditions. The results of these models will be factored into future inspection/pipe replacement plans consistent with the current FAC Program requirements. p4-2	X		implementation date
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 the actions required upon loss of the LEFM Checkplus instrument p4-2	X		implementation date

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 p4-3	X		implementation date
The COLSS licensed power monitoring algorithm will be modified and this will be identified and included in the training program. P4-3	X		implementation date
Classroom and simulator training will be provided on all changes that affect operator performance caused by this power uprate. Changes to the simulator will be made consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be validated in accordance with ANSI/ANS 3.5-1985, section 5.4.1, "Simulator Performance Testing." All control room and plant process computer system changes a s a result of the power uprate will be completed. P4-3	X		Implementation date