CP&L
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U. S. Nuclear Regulatory Commission
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BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
SUBMITTAL OF NON-PROPRIETARY SAFETY ANALYSIS REPORT
EXTENDED POWER UPRATE

Ladies and Gentlemen:

On August 9, 2001, Carolina Power & Light Company (CP&L) submitted a license amendment request (Serial: BSEP 01-0086) to increase the maximum power level, for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, from 2558 megawatts thermal (MWt) to 2923 MWt. BSEP 01-0086 included NEDC-33039P, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Extended Power Uprate," dated August 2001.

The purpose of this letter is to provide a non-proprietary version of NEDC-33039P. The Enclosure to this letter contains NEDO-33039, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Extended Power Uprate," dated August 2001. NEDO-33039 is suitable for public disclosure.

Please refer any questions regarding this submittal to Mr. Leonard R. Beller, Supervisor - Licensing/Regulatory Programs, at (910) 457-2073.

Sincerely,

Leonard R. Beller
FOR

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Enclosure:

NEDO-33039, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Extended Power Uprate," dated August 2001

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ENCLOSURE

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NEDO-33039
Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2
Extended Power Uprate
dated August 2001

GE Nuclear Energy
175 Curtner Ave., San Jose, CA 95125

NEDO-33039
Revision 0
DRF A22-00113-53
Class I
August 2001

SAFETY ANALYSIS REPORT
FOR
BRUNSWICK STEAM ELECTRIC PLANT
UNITS 1 AND 2
EXTENDED POWER UPRATE

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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ACRONYMS AND ABBREVIATIONS

Term	Definition
AC	Alternating Current
ADS	Automatic Depressurization System
APRM	Average Power Range Monitor
ARTS	APRM/RBM/Technical Specifications
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
B&PV	Boiler and Pressure Vessel
BHP	Brake Horsepower
BOP	Balance-of-plant
BSEP	Brunswick Steam Electric Plant
BWR	Boiling Water Reactor
CACS	Containment Atmosphere Containment System
CAD	Containment Atmosphere Dilution
CAM	Containment Atmosphere Monitoring
CFR	Code of Federal Regulations
CLTP	Current Licensed Thermal Power
CO	Condensation Oscillation
COLR	Core Operating Limits Report
CPD	Condensate Polishing Demineralizer
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CS	Core Spray
DBA	Design Basis Accident
DC	Direct Current
DFG	Diode Function Generator
DOR	Division of Operating Reactors
ECCS	Emergency Core Cooling System

Term	Definition
EHC	Electro-hydraulic Control
EOC	End-of-cycle
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Environmental Qualification
ESF	Engineered Safety Features
ESW	Emergency Service Water
FAC	Flow Assisted Corrosion
FPC	Fuel Pool Cooling
FPCC	Fuel Pool Cooling and Cleanup
FCS	Feedwater Control System
FES	Final Environmental Statement
FHA	Fire Hazards Analysis
GDC	General Design Criterion
GE	General Electric Company
GL	Generic Letter
GSW	General Service Water
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Adsorber
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilating and Air Conditioning
HWC	Hydrogen Water Chemistry
ILBA	Instrument Line Break Accident
IPE	Individual Plant Evaluation
IPEEE	Individual Plant Examination – External Events
IRM	Intermediate Range Monitor
LCO	Limiting Condition for Operation
LLRPSF	Low Level Radwaste Processing Storage Facility
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater Flow
LPCI	Low Pressure Coolant Injection
LPSP	Low Power Setpoint

Term	Definition
LTP	Long Term Program
LTR	Licensing Topical Report
MCPR	Minimum Critical Power Ratio
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MG	Motor-Generator
Mlb/hr	Million Pounds Per Hour
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MSLBA	Main Steam Line Break Accident
MWe	Megawatt-electric
MWt	Megawatt-thermal
NEI	Nuclear Energy Institute
NPDES	National Pollutant Discharge Elimination System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NWC	Normal Water Chemistry
OOS	Out of Service
OLTP	Original Licensed Thermal Power
PCS	Pressure Control System
PCT	Peak Clad Temperature
PSA	Probabilistic Safety Assessment
PUSAR	Power Uprate Safety Analysis Report
QA	Quality Assurance
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RHR	Residual Heat Removal

Term	Definition
RHRSW	Residual Heat Removal Service Water
RIPD	Reactor Internal Pressure Difference
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RRS	Reactor Recirculation System
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
SAT	Startup Auxiliary Transformer
SBO	Station Blackout
SFP	Spent Fuel Pool
SFU	Standby Filter Unit
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejectors
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-loop Operation
SRM	Source Range Monitor
SRV	Safety/Relief Valve
SRVDL	Safety/Relief Valve Discharge Line
SSC	Structures, Systems, and Components
STA	Spurious Trip Avoidance
SV	Safety Valve
TAF	Top of Active Fuel
TBCCW	Turbine Building Closed Cooling Water
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TLO	Two (recirculation) Loop Operation
TSC	Technical Support Center

Term	Definition
TSCR	Technical Specification Change Request
TSV	Turbine Stop Valve
UAT	Unit Auxiliary Transformer
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink

EXECUTIVE SUMMARY

This report summarizes the results of all significant safety evaluations performed that justify extending the licensed thermal power at the Brunswick Steam Electric Plant, Units 1 and 2 (BSEP 1 and 2) to 2923 MWt. The requested license power level is 20% above the Original Licensed Thermal Power (OLTP) of 2436 MWt.

Uprating the power level of nuclear power plants can be done safely within certain plant-specific limits and is a cost effective way to increase installed electrical generating capacity. An increase in electrical output of a General Electric (GE) Boiling Water Reactor (BWR) plant is accomplished primarily by generation and supply of higher steam flow to the turbine generator. BSEP 1 and 2, as originally licensed, has an as-designed equipment and system capability to accommodate steam flow rates at least 5% above the current rating. Also, the plant has sufficient design margins to allow the plant to be safely uprated up to 120% of its OLTP.

Detailed evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accident analyses and previous licensing evaluations were performed.

This report supports the conclusion that this EPU can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without exceeding any existing regulatory limits applicable to the plant. The environmental evaluation demonstrated that the EPU does not involve environmental effects that differ significantly from those previously evaluated for the presently authorized Rated Thermal Power (RTP) level. Where environmental impacts differ from those previously evaluated, these effects have been shown to be insignificant. The EPU described herein involves no significant hazard consideration.

1 INTRODUCTION AND SUMMARY

1.1 Introduction

Uprating the power level of nuclear power plants can be done safely within certain plant-specific limits. Most General Electric (GE) Boiling Water Reactor (BWR) plants, including the Brunswick Steam Electric Plant, Units 1 and 2 (BSEP 1 and 2), have the capability and margins for a power uprate of up to 20% without major Nuclear Steam Supply System (NSSS) hardware modifications.

The evaluation presented in this report justifies an extended power uprate (EPU) to 2923 MWt, which corresponds to 120% of the Original Licensed Thermal Power (OLTP) level of 2436 MWt. The generic criteria, process, and scope of work required to provide sufficient information for use by the Nuclear Regulatory Commission (NRC) to grant approval to specific applications for increases in the authorized thermal power levels for GE BWRs are contained in ELTR1 (Reference 1). This report follows the NRC-approved generic process requirements contained in ELTR1.

1.2 Purpose And Approach

An increase in electrical output of a BWR is accomplished primarily by generation and supply of higher steam flow to the turbine generator. Most BWRs, as originally licensed, have an as-designed equipment and system capability to accommodate steam flow rates above the original rating. In addition, continuing improvements in the analytical techniques and computer codes, plant performance feedback/operating experience, and implementation of improvements in fuel designs have resulted in a significant increase in the design and operating margins between the calculated safety analyses results and the licensing limits. These available differences in calculational results, combined with the as-designed excess equipment, system, and component capabilities: (1) have allowed numerous BWRs to increase their thermal power ratings by 5% without any NSSS hardware modification, and (2) provide for power increases to 20% with some hardware modifications. These power increases involve no significant increase in the hazards presented by the plants as approved by the NRC at the original license stage.

BSEP 1 and 2 are currently licensed for a 100% RTP level of 2558 MWt. The safety analyses of design basis accidents (DBAs) and operational transients are based on a power level 102% above the proposed EPU RTP level of 2923 MWt, unless the 2% power factor is already accounted for in the analysis methods.

The EPU analysis basis ensures that the power-dependent safety margin prescribed by the Code of Federal Regulations (CFR) is maintained by meeting the appropriate regulatory criteria. Either NRC-approved or industry-accepted computer codes and calculational techniques are used to demonstrate meeting the applicable regulatory acceptance criteria.

The planned approach to achieving the higher power level consists of: (1) an increase in the core thermal power to create increased steam flow to the turbine, (2) a corresponding increase in the Feedwater system flow, (3) no increase in either maximum core flow or reactor dome pressure, and (4) reactor operation primarily along an extension of the standard Maximum Extended Load Line Limit Analysis (MELLLA) rod/flow control lines. Plant-unique evaluations were based on a review of plant design and operating data, as applicable, to confirm excess design capabilities, and, if necessary, identify any items which may require modifications associated with the EPU. For some items, bounding analyses and evaluations demonstrate plant operability and safety. The scope and depth of the evaluation results provided herein were established based on the generic BWR EPU guidelines and unique features of the plant. The results of the applicable evaluations presented in this report were found to be acceptable.

1.3 EPU Plant Operating Conditions

The thermal-hydraulic performance of a BWR reactor core is characterized by the operating power, the operating pressure, the total core flow, and the coolant thermodynamic state. The rated values of these parameters are used to establish the steady-state operating conditions and as initial and boundary conditions for the required safety analyses. They are determined by performing heat (energy) balance calculations for the Reactor system at the EPU conditions.

The EPU heat balance was determined such that the core thermal power is 120% of the OLTP and the steam flow from the vessel was increased to approximately 14.3% above the current value. The reactor heat balance is coordinated with the turbine heat balance. Figure 1-1 shows the EPU heat balance at 100% of EPU RTP and 100% rated core flow. Table 1-1 provides a summary of the reactor thermal-hydraulic parameters for the current rated condition and the EPU condition.

1.4 Summary And Conclusions

This report supports the conclusion that this EPU can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without exceeding any existing regulatory limits applicable to the plant. The environmental

evaluation demonstrated that the EPU does not involve environmental effects that differ significantly from those previously evaluated for the presently authorized RTP level. Where environmental impacts differ from those previously evaluated, these effects have been shown to be insignificant. The EPU described herein involves no significant hazard consideration.

Table 1-1

Current and EPU Plant Operating Conditions

Parameter	Current RTP Value [1]	EPU RTP Value
Thermal Power (MWt)	2558	2923
Vessel Steam Flow [2] (Mlb/hr)	11.089	12.781
Full Power Core Flow Range		
Mlb/hr (Unit 1)	62.4 to 80.3	76.2 to 80.5
% Rated (Unit 1)	81 to 104.3	99 to 104.5
Mlb/hr (Unit 2)	62.4 to 80.5	76.2 to 80.5
% Rated (Unit 2)	81 to 104.5	99 to 104.5
Dome Pressure (psia)	1045	No change
Dome Temperature (°F)	549.9	No change
Turbine Stop Valve Inlet Pressure (psia)	991	973
Full Power Feedwater		
Flow [2] (Mlb/hr)	11.063	12.755
Temperature Range (°F)	315.0 to 425.3	321.1 to 431.4
Core Inlet Enthalpy [3] (Btu/lb)	529.8	528.2

Notes:

(1) Some values for the current RTP level were recalculated with inputs and assumptions consistent with those used for the EPU heat balance. As such, these values may not be the same as currently shown in the UFSAR.

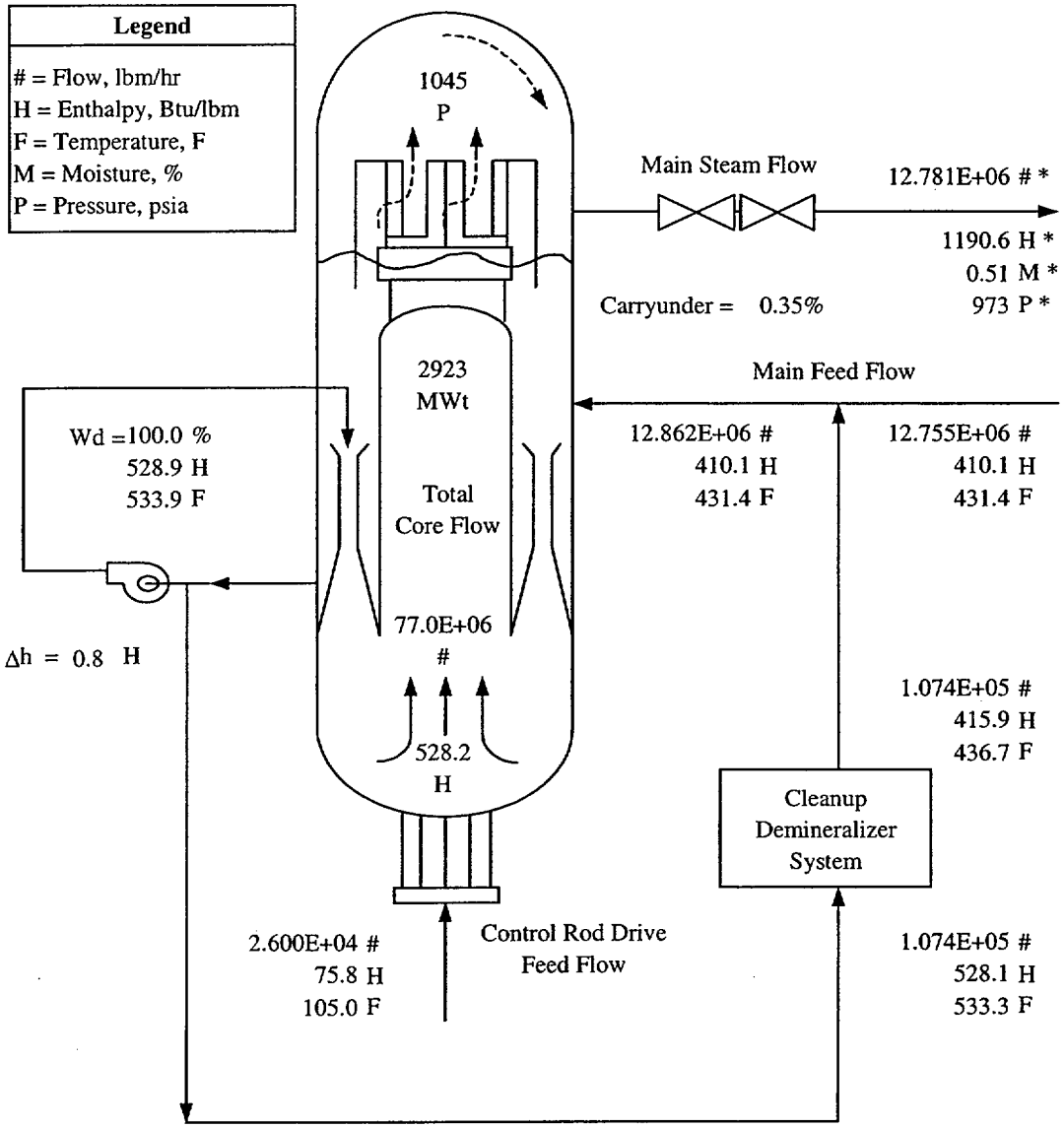
(2) At normal feedwater heating

(3) At the 100% core flow condition

Performance improvement features and/or equipment out-of-service (OOS) included in the EPU evaluations are:

1) Maximum Extended Operating Domain (MEOD) / MELLLA
2) Single-loop Operation (SLO)
3) One Safety Relief Valve (SRV) OOS
4) Two Automatic Depressurization System (ADS) valves OOS

5) Average Power Range Monitor (APRM) / Rod Block Monitor (RBM) / Technical Specifications (ARTS)
6) Turbine Bypass Valve (TBV) OOS
7) Feedwater Heater (FWH) OOS
8) Main Steam Isolation Valve (MSIV) OOS
9) Operation with either Startup Auxiliary Transformer (SAT) or Unit Auxiliary Transformer (UAT) for Recirculation Pump Power Source



* Conditions at upstream side of TSV

Core Thermal Power	2923.0
Pump Heating	7.8
Cleanup Losses	-3.5
Other System Losses	-1.1
Turbine Cycle Use	2926.2 MWt

Figure 1-1

EPU Heat Balance – Nominal
(@ 100% Power and 100% Core Flow)

2 REACTOR CORE AND FUEL PERFORMANCE

2.1 Fuel Design and Operation

At the OLTP or the EPU conditions, all fuel and core design limits continue to be met by planned deployment of fuel enrichment and burnable poison, and supplemented by core management control rod pattern and/or core flow adjustments. Revised loading patterns, larger batch sizes, and new fuel designs are used to provide additional operating flexibility and maintain fuel cycle length.

2.2 Thermal Limits Assessment

Operating limits ensure that regulatory and/or safety limits are not exceeded for a range of postulated events [e.g., transients, loss-of-coolant accidents (LOCA)]. Cycle-specific core configurations, evaluated for each reload, confirm EPU RTP capability and establish or confirm cycle-specific limits, as is currently the practice. The evaluation of thermal limits for the EPU core shows that the current thermal margin design limits can be maintained.

2.3 Reactivity Characteristics

All minimum shutdown margin requirements apply to cold (≤ 20°C) conditions, and are maintained without change. The Technical Specifications cold shutdown margin requirements are not affected. Operation at higher power could reduce the hot excess reactivity during the cycle. This loss of reactivity does not affect safety, and is not expected to significantly affect the ability to manage the power distribution through the cycle to achieve the target power level.

The EPU power-flow operating map (Figure 2-1) includes the operating domain changes for the EPU and the plant performance improvement features currently allowed for in the Updated Final Safety Analysis Report (UFSAR), core fuel reload evaluations, and/or the Technical Specifications. The maximum thermal operating power and maximum core flow shown on Figure 2-1, correspond to the EPU RTP. Figure 2-1 shows the current maximum licensed rod line and the proposed maximum rod line for EPU on an absolute power basis.

2.4 Stability

BSEP 1 and 2 are currently operating under the requirements of reactor stability Long-Term Solution Enhanced Option I-A (E1A), and are in the process of implementing reactor stability Long-Term Solution Option III. The EPU is scheduled to be implemented in BSEP 1 Cycle 14 and BSEP 2 Cycle 16, and the Oscillation Power Range Monitor (OPRM) system is also

scheduled to be armed for those cycles. Because the Interim Corrective Actions (ICAs) are used as a backup solution if the OPRM system fails, the effect of the EPU is addressed on both the ICAs (Reference 2) and on the stability Option III solution (Reference 3).

To ensure adequate level of protection against the occurrence of a thermal-hydraulic instability, the instability exclusion region boundaries are unchanged with respect to absolute power level (MWt).

The Option III solution monitors OPRM signals to determine when a reactor scram is required. The OPRM system may only cause a scram when plant operation is in the Option III OPRM Trip Enabled Region. The OPRM Trip Enabled Region will be defined in the Technical Specifications and plant procedures, and will be incorporated on the BSEP power/flow operating map. The OPRM Trip Enabled Region was modified for EPU operation to maintain the pre-EPU absolute power and flow coordinates. The stability-based Operating Limit Minimum Critical Power Ratio (OLMCPR) associated with the OPRM setpoint assures that the MCPR safety limit is not violated during an instability event.

2.5 Reactivity Control

The Control Rod Drive (CRD) system introduces changes in core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core.

Because there is no increase in the vessel operating pressure, CRD scram performance and CRD mechanism structural and functional integrity are not affected by the EPU.

The components of the CRD mechanism, which form part of the primary pressure boundary, have been designed in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code, Section III. The EPU engineering analyses show that all stresses and fatigue usage factors remain within their original design allowable values.

Based on the above, the CRD system is acceptable for the EPU.

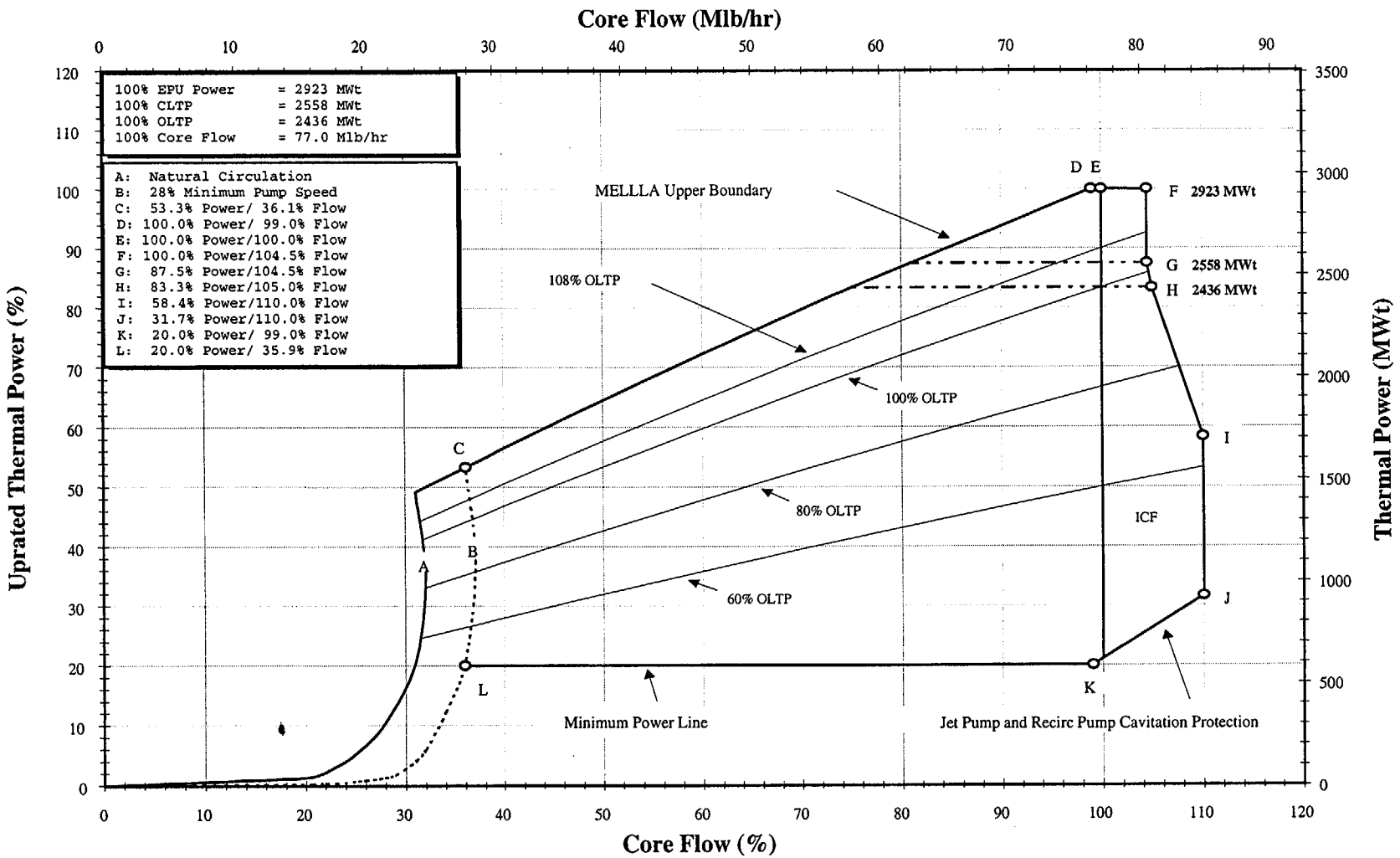


Figure 2-1

Power-Flow Operating Map For EPU

3 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

3.1 Nuclear System Pressure Relief

The purpose of the nuclear system pressure relief is to prevent overpressurization of the nuclear system during abnormal operational transients. The plant safety relief valves (SRVs) with scram provide this protection. The SRV setpoints are not changed with the EPU, because the maximum operating dome pressure is not changed.

3.2 Reactor Overpressure Protection

The design pressure of the reactor vessel and reactor pressure coolant boundary remains at 1250 psig. The acceptance limit for pressurization events is the ASME code allowable peak pressure of 1375 psig (110% of design value). The limiting pressurization event remains the MSIV closure with flux scram. Starting from EPU RTP conditions, the peak calculated reactor pressure vessel (RPV) pressure remains below the 1375 psig ASME limit and reactor steam dome pressure remains below the Technical Specification 1325 psig Safety Limit. Therefore, there is no decrease in margin of safety.

3.3 Reactor Vessel And Internals

Evaluations of the reactor vessel and vessel internals concluded that the corresponding peak vessel loads and fluence conditions resulting from this EPU were within the existing design bases of these structures.

The estimated fluence for EPU conditions was conservatively increased above the UFSAR end-of-life value. Therefore, the higher fluence was used to evaluate the vessel against the requirements of 10 CFR 50 Appendix G. The vessel remains in compliance with the regulatory requirements during EPU conditions.

With regards to the structural integrity of the reactor vessel components, because there are no changes in the design conditions due to the EPU, the design stresses are unchanged and the ASME Code requirements applicable to BSEP are still met. Further, because there is no pressure increase and only minor changes to some temperatures and flows, the analysis results for normal, upset, emergency, and faulted conditions show that all components meet their ASME Code requirements.

The increase in core average power results in higher core loads and reactor internal pressure differences (RIPDs) due to the higher core exit steam quality. The RIPDs were re-calculated for

normal steady-state operation, upset, and faulted conditions for all major reactor internal components and determined to be acceptable.

The results of a vibration evaluation show that operation up to 102% EPU RTP and 105% of rated core flow is possible without any detrimental effects on the safety-related reactor internal components.

The expected performance of the steam separators and dryer was evaluated to ensure that the quality of the steam leaving the reactor pressure vessel continues to meet existing operational criteria at the EPU conditions. The results of the evaluation demonstrate that the steam separator-dryer performance remains acceptable at the EPU conditions.

3.4 Reactor Recirculation System

An evaluation of the Reactor Recirculation System (RRS) performance concluded that the existing design margin of the RRS is well within the slight changes in system temperature and pressure resulting from EPU.

3.5 Reactor Coolant Pressure Boundary Piping

The effects of EPU were evaluated for the reactor coolant piping systems which are part of the primary reactor coolant pressure boundary (RCPB) and which could be affected by an EPU-related increase in flow or operating temperature. These evaluations concluded that EPU does not have an adverse effect on the primary piping systems design. The slight increase in temperature associated with the EPU that affects piping and piping support loads does not result in load limits being exceeded.

The Recirculation system components are made of stainless steel, and system flow increase due to the EPU is minor. Stainless steel piping is not susceptible to flow-accelerated corrosion (FAC).

The Main Steam and associated piping systems and Feedwater system piping are made of carbon steel, which can be affected by FAC (erosion/corrosion). The integrity of high energy piping systems is assured by proper design in accordance with the applicable Codes and Standards. The plant has an established program for monitoring pipe wall thinning in single-phase and two-phase high-energy carbon steel piping. Other RCPB piping systems [RPV head vent, bottom head drain, and portions of the High Pressure Coolant Injection (HPCI), Reactor Core Isolation

Cooling (RCIC), and Reactor Water Cleanup (RWCU) systems] affected by FAC are also included in this program.

EPU operation results in some changes to parameters affecting flow-induced erosion/corrosion in those systems associated with the turbine cycle (e.g., Condensate, Feedwater, Main Steam). The evaluation of and inspection for flow-induced erosion/corrosion in Balance-of-Plant (BOP) piping systems that is affected by FAC is addressed by compliance with NRC Generic Letter 89-08, "Erosion/Corrosion in Piping." EPU evaluations have confirmed that the EPU has no significant effect on flow-induced erosion/corrosion.

3.6 Main Steam Line Flow Restrictors

An evaluation of the main steam line flow restrictors concluded that the existing design margin of the flow restrictors is well within the slight changes in conditions resulting from EPU.

3.7 Main Steam Isolation Valves

The MSIVs are part of the RCPB and must be able to close within specific limits at all design and operating conditions upon receipt of a closure signal. The MSIVs have been evaluated and are acceptable for EPU operation.

3.8 Reactor Core Isolation Cooling System

The RCIC system provides core cooling in the event of a transient where the RPV is isolated from the main condenser concurrent with the loss of all feedwater flow. For EPU, the reactor dome pressure and the SRV setpoints remain unchanged. Consequently, there is no change to the RCIC high-pressure injection process parameters and no change to the overspeed trip margins. The existing RCIC capacity is adequate to maintain reactor water level above the top of the active fuel (TAF) for the Loss of Feedwater Flow (LOFW) transient as described in Section 9.1.

3.9 Residual Heat Removal System

The Residual Heat Removal (RHR) system is designed to restore and maintain the coolant inventory in the reactor vessel and to remove sensible and decay heat from the primary system and containment following reactor shutdown for both normal and post accident conditions. Evaluations indicate that the implementation of EPU does not prevent any of the RHR modes from performing their intended functions.

3.10 Reactor Water Cleanup System

The RWCU system is designed to remove solid and dissolved impurities from recirculated reactor coolant, thereby reducing the concentration of radioactive and corrosive species in the reactor coolant. Operation of the plant at the EPU RTP level does not increase the temperature or the pressure within the RWCU system nor is the radioactive content of the reactor water significantly increased. EPU results in a slight increase in the reactor water conductivity because of the increase in feedwater flow. However, the reactor water conductivity limits will be met. Therefore, implementation of the EPU does not prevent the system from performing its intended function.

3.11 Balance-Of-Plant Piping

This section addresses the adequacy of the BOP piping design outside the RCPB for operation at the EPU conditions.

Large bore and small bore piping and supports not addressed in Section 3.5 were evaluated for acceptability at the EPU conditions, and shown to be adequate as currently designed. The evaluation of the BOP piping and supports was performed in a manner similar to the evaluation of RCPB piping systems and supports (Section 3.5), using applicable B31.1 Power Piping Code equations. The original Codes of record (as referenced in the appropriate calculations), Code allowable and analytical techniques were used, and no new assumptions were introduced.

Operation at the proposed EPU conditions increases pipe stresses due to slightly higher operating temperatures and flow rates internal to the pipes. For all systems, the maximum stress levels and fatigue analysis results were reviewed based on specific increases in temperature and flow rate and were found to meet the appropriate code criteria for the EPU conditions.

Operation at EPU conditions causes a slight increase in the pipe support loadings due to increases in the temperature of the affected piping systems. However, when considering the loading combination with other loads that are not affected by EPU, such as seismic and deadweight, the overall combined support load increase is insignificant. There is adequate margin between the original design stresses and code limits of the supports to accommodate the load increase within the appropriate code criteria. Therefore, the design of the BOP piping systems is adequate to accommodate the EPU.

EPU operation results in some changes to parameters affecting flow-induced erosion/corrosion in those systems associated with the turbine cycle (e.g., Condensate, Feedwater, Main Steam). The

evaluation of and inspection for flow-induced erosion/corrosion in BOP piping systems is addressed by compliance with NRC Generic Letter 89-08, "Erosion/Corrosion in Piping." Evaluations have confirmed that the EPU has no significant effect on flow-induced erosion/corrosion. The affected systems are currently monitored by the plant Erosion/Corrosion Program. Continued monitoring of the systems provides a high level of confidence in the integrity of potentially susceptible high energy piping systems. Appropriate changes to piping inspection frequency will be implemented to ensure adequate margin exists for those systems with changing process conditions. This program provides assurance that the EPU has no adverse effect on high energy piping systems potentially susceptible to pipe wall thinning due to erosion/corrosion.

4 ENGINEERED SAFETY FEATURES

4.1 Containment System Performance

The UFSAR provides the containment responses to various postulated accidents that validate the design basis for the containment. Operation during EPU changes some of the conditions for the containment analyses. The containment pressure and temperature responses have been reanalyzed to demonstrate the plant's capability to operate with the EPU. The results of the analyses are as follows:

- The calculated peak bulk suppression pool temperature remains below the design temperature.
- The calculated drywell airspace temperature remains below the drywell shell design temperature.
- The calculated drywell pressure remains well below the containment design pressure.
- The effect of EPU on net positive suction head (NPSH) for pumps taking suction from the suppression pool was evaluated. The NPSH margin for the RHR and core spray (CS) pumps is negative at the peak suppression pool temperature. Therefore, containment overpressure must be credited to ensure adequate NPSH.

The LOCA containment dynamic loads analysis for the EPU is based primarily on the short-term DBA-LOCA analyses. The LOCA dynamic loads with the EPU include pool swell, condensation oscillation (CO) and chugging. For Mark I plants like BSEP 1 and 2, the vent thrust loads are also evaluated.

The short-term DBA-LOCA containment responses are within the range of test conditions used to define the pool swell and CO loads for the plant. The containment responses with the EPU in which chugging would occur are within the conditions used to define the chugging loads. Therefore, the existing definitions for the DBA-LOCA dynamic loads remain applicable at EPU conditions.

The SRV discharge loads include SRV discharge line (SRVDL) loads, suppression pool boundary pressure loads and drag loads on submerged structures. For initial SRV actuation, the only parameter that can affect the SRV loads is the SRV opening setpoint pressure. However, this EPU does not include an increase in the SRV opening setpoint pressures. Therefore, the SRV discharge loads due to first actuation remain bounded by the existing load definition.

The effect of EPU on subsequent actuation loads due to changes in the SRVDL water level and time between actuations was also evaluated. The existing load definition for SRV re-actuation also remains applicable to EPU conditions.

The systems designed for containment isolation are not affected by the EPU. The capability of the actuation devices to perform during normal operation and under post-accident conditions has been determined to be acceptable.

All motor-operated valves (MOVs) included in the Generic Letter (GL) 89-10 Program were evaluated for the effects of the EPU, including potential locking and thermal binding (GL 95-07). If specific valves require calculation revisions, actuator adjustments and/or physical changes to ensure satisfactory performance, then these upgrades and any other field adjustments or modifications will be performed prior to EPU operation.

The plant's past response to GL 96-06 was also reviewed for the EPU post accident conditions. CP&L is participating in an industry collaborative project with the Electric Power Research Institute (EPRI) and the Nuclear Energy Institute (NEI) to develop a generic technical basis to address the water hammer issues. CP&L committed to providing an update of intended actions with respect to GL 96-06 after the NRC approves the EPRI/NEI generic technical basis. Post-EPU containment temperatures and pressures will be used in any technical analyses developed to support the GL 96-06 evaluation.

4.2 Emergency Core Cooling Systems

The Emergency Core Cooling Systems (ECCS) are designed to provide protection against hypothetical LOCAs caused by ruptures in the primary system piping. The functional capability of each system was determined to be acceptable for the EPU.

Originally, the HPCI system was primarily for the mitigation of small break LOCAs where the depressurization function (ADS / SRVs) was assumed to fail. For BSEP, the depressurization function is fully redundant, and no accident mitigation credit is taken for the HPCI system. The primary remaining purpose of the HPCI system is to maintain reactor level above the TAF and prevent ADS actuation for line breaks up to 1" in diameter.

The Low Pressure Coolant Injection (LPCI) mode of the RHR system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, the LPCI mode is required to provide adequate core cooling for all LOCA events. EPU did not increase the calculated peak clad temperature (PCT) following a postulated LOCA. The evaluation of the

LPCI system indicates that the existing LPCI mode performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for the EPU conditions.

The CS system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, the CS system is designed to provide adequate core cooling for any applicable LOCA event. EPU did not increase the calculated PCT following a postulated LOCA. The evaluation of the CS system indicates that its existing performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement for the EPU conditions.

The ADS uses SRVs to reduce reactor pressure following a small break LOCA, when it is assumed that the high pressure ECCS has failed. This function allows LPCI and CS to inject coolant into the vessel. The evaluation of small break LOCAs demonstrates that ADS capacity is adequate when operating at the EPU conditions. The ADS initiation logic and ADS valve control are not affected. Thus, ADS is adequate for the EPU conditions.

Therefore, the ECCS performance under all LOCA conditions, and their analysis models, satisfy the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K.

4.3 Main Control Room Atmosphere Control System

The main control room atmosphere control system is not significantly affected by the EPU and control room operator doses remain well below regulatory limits.

4.4 Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) is designed to minimize offsite and control room doses during venting and purging of the primary and secondary containment atmosphere under accident or abnormal conditions. The capacity of the SGTS was selected to maintain the secondary containment at a slight negative pressure. This capability is not affected by the EPU.

The charcoal filter bed removal efficiency for radioiodine is unaffected by the EPU. As a result of the EPU and application of Alternative Source Term (AST) derived from Regulatory Guide (RG) 1.183 (see Section 9.2), the post-DBA-LOCA total iodine loading is 0.003 mg/gm of charcoal at the EPU conditions, which is well below the RG 1.52 value. The system therefore contains sufficient charcoal to ensure iodine removal efficiencies greater than the current design requirement.

4.5 Post-LOCA Combustible Gas Control

The Combustible Gas Control system is designed to maintain the oxygen concentrations of the drywell and containment atmospheres below the lower flammability limit following a hypothetical LOCA. The post-LOCA production of hydrogen and oxygen by radiolysis increases proportionally with power level. The increase in radiolysis due to the EPU has a minor effect on the time available to start the system before reaching procedurally controlled limits, but does not affect the ability of the system to maintain oxygen below the lower flammability limit of 5% by volume as specified in Safety Guide 7. The required start time for the containment atmosphere dilution (CAD) system decreases from 6.2 days to 5.3 days for the EPU. This reduction in required CAD initiation time does not affect the ability of the operators to respond.

The on-site nitrogen storage volume is adequate to maintain the containment atmosphere at or below the 5% oxygen flammability limit for 29 days post-LOCA, as compared to a minimum of 30 days for current conditions. This change is not significant and allows adequate time to replenish the storage tank from off-site sources. Analysis of the containment pressure buildup as a result of continuing CAD operation shows that the containment repressurization limit of 31 psig (50% of the design pressure) is not exceeded until 29 days after the LOCA. More realistic analyses using typical initial inerting levels of approximately 1% oxygen and containment leakage below the allowable 0.5% per day, versus the 4% oxygen Technical Specification limit and zero containment leakage, extend both the nitrogen supply availability and the approach to the repressurization limit to over 30 days.

5 INSTRUMENTATION AND CONTROL

5.1 Nuclear Steam Supply System

This EPU involves no increase in reactor pressure, and the pressure-dependent setpoints do not require modification. However, increases in core thermal power and steam flow affect some instrument setpoints.

The APRM power signals will be rescaled to the 2923 MWt power level, such that the indications read 100% at the new licensed power level.

EPU has little effect on the intermediate range monitor (IRM) overlap with the source range monitors (SRMs) and the APRMs. Using normal plant surveillance procedures, the IRMs may be adjusted, as required, so that overlap with the SRMs and APRMs remains adequate. No change is needed in the APRM downscale setting.

The Rod Worth Minimizer (RWM) does not perform a safety-related function. The function of the RWM is to support the operator by enforcing rod patterns until reactor power has reached appropriate levels. Specifically, the RWM satisfies Criteria 3 of 10 CFR 50.36 and functions to limit the local power in the core to maintain the effects of the postulated Control Rod Drop Accident (CRDA) while reactor power is < 10% of Current Licensed Thermal Power (CLTP) (< 8.75% EPU RTP).

The determination of instrument setpoints is based on plant operating experience, conservative licensing analyses, and/or (limiting) design/operating values. Each setpoint is selected with sufficient margin between the actual trip setting and the value used in the safety analysis [i.e., the analytical limit (AL)] to allow for instrument accuracy, calibration, and drift. Sufficient margin is provided wherever possible between the actual trip setting and the normal operating limit to ensure timely actuation of the necessary safety functions while avoiding spurious trips wherever possible during EPU operation.

The following instrument analytical limits remain unchanged due to implementation of the EPU:

- Reactor vessel high-pressure scram
- Anticipated transient without scram (ATWS) recirculation pump trip (RPT) high pressure trip
- SRV setpoints

- Main steam high flow isolation (in percent of rated steam flow)
- The APRM simulated thermal power (STP) scram AL remains unchanged, however, the flow-biased scram AL is changed as identified below.
- The Rod Block Monitor (RBM) system has three upscale trip levels, which are based on three thermal power level ranges. These power levels in terms of percent of rated thermal power are not changed.
- Main steam line high radiation isolation
- Low steam line pressure MSIV closure
- Reactor water level instruments
- Main steam line tunnel high temperature isolations
- Low steam line MSIV isolation
- RCIC steam line high flow isolation
- HPCI steam line high flow isolation

The following instrument ALs are changed due to implementation of the EPU:

- The APRM flow-biased STP scram is redefined to reflect the change in the maximum allowable load line region.
- The turbine stop valve closure and turbine control valve fast closure scram bypass AL is reduced by the ratio of the power increase. However, the new AL does not change in terms of absolute power.
- The RWM AL is also reduced by the ratio of the power increase.

5.2 Balance-Of-Plant

Operation of the plant at the EPU RTP level has minimal effect on the BOP system instrumentation and control devices. Any required changes will be performed prior to operation at the EPU RTP.

The Pressure Control System (PCS) provides fast and stable response to system disturbances related to pressure and steam flow changes to control reactor pressure within its normal operating range. The PCS consists of the pressure regulation system, the turbine-generator electro-hydraulic control (EHC) system and the steam bypass valve system. The main turbine speed/load control function is performed by the EHC system. The steam pressure control function is performed by the pressure regulation system, through manipulation of the turbine control valves and the bypass valves. With modifications such as changes to the high-pressure turbine and adjustments to the turbine control valve diode function generators (DFGs) in the EHC, sufficient pressure control range would be available to control system disturbances at the EPU conditions. Thus, the existing main turbine-generator EHC, the pressure regulation system, and the steam bypass control system are adequate for the EPU conditions. Specific PCS tests will be performed during the power ascension phase.

The turbine-generator EHC system was reviewed for the increase in core thermal power and the associated increase in rated steam flow. New TCV DFG tuning and updating of the characteristic TCV tuning parameter curve are necessary for the control systems to perform normally at the EPU conditions. The control systems are expected to perform normally for EPU RTP operation.

No modifications to the turbine control valves or the turbine bypass valves are required for operation at the EPU throttle conditions. Normal manual operator controls will be used in conjunction with the associated operating procedures. Confirmation testing will be performed during power ascension.

The feedwater control system controls reactor water level during normal operations. The control system itself is adjusted to provide acceptable operating response on the basis of unit behavior. It has been set up successfully to cover the current power range using startup and periodic testing. No changes in the operating water level or water level trip setpoints are required for the EPU. For the EPU, the feedwater flow control system device settings have sufficient adjustment ranges to ensure satisfactory operation. However, the feedwater flow transmitters and associated components will be re-calibrated for proper operation at EPU conditions. This will be confirmed by performing unit tests during the power ascension to the EPU conditions.

The instrument setpoints associated with system leak detection have been evaluated with respect to the slightly higher operating steam flow and feedwater temperature for the EPU. There is no significant effect on any leak detection system due to the EPU.

6 ELECTRICAL POWER AND AUXILIARY SYSTEMS

6.1 Alternating Current Power

The existing off-site electrical equipment was determined to be adequate for operation with the EPU-related electrical output, as shown in Table 6-1. The review concluded the following.

- The BSEP 1 isolated phase bus cooling will be modified prior to exceeding the CLTP to handle the additional loads associated with EPU. The BSEP 2 isolated phase bus cooling will be modified prior to exceeding 113% OLTP.
- The BSEP 1 and 2 Main Transformers are capable of continuously carrying the maximum generator outputs expected up to 113% OLTP maximum. However, the transformers will be replaced prior to exceeding 113% OLTP.
- The existing Generator and Main Transformer Protective Relaying scheme will require minor modifications to ensure reliable operation prior to achieving full EPU RTP. This equipment, however, is adequate as designed for reliable operation at 113% OLTP.

A grid stability analysis has been performed, considering the increase in electrical output, to demonstrate conformance to General Design Criteria (GDC) 17 (10 CFR 50, Appendix A) with respect to stability. This analysis determined that several modifications and procedure changes should be implemented to ensure grid stability and reliability. Out-of-Step Protective Relays should be installed prior to exceeding CLTP to protect the main generator and minimize loss of offsite power. Power system stabilizers should be installed on BSEP 1 prior to exceeding 111% OLTP and on BSEP 2 prior to exceeding CLTP to provide adequate damping of post-transient oscillations. In addition, during key line outages, maximum generator output may be limited to maintain adequate damping of oscillations; the procedural controls for limiting generator output will be established prior to exceeding CLTP.

The onsite power distribution system consists of transformers, buses, and switchgear. Alternating Current (AC) power to the distribution system is provided from the transmission system or from onsite diesel generators. Station batteries provide Direct Current (DC) power to the distribution system.

Station loads under normal operation/distribution conditions are computed based on equipment nameplate data with conservative demand factors applied. The only identifiable change in electrical demand is associated with load increases for Recirculation Pumps, Condensate Pumps, Condensate Booster Pumps, Stator Water Cooling Pumps, Main Transformer controls/cooling

equipment, Isolated-Phase Bus cooling equipment, and a new Condensate Cooling system. The Condensate, Condensate Booster and Stator Water Cooling pumps require larger motors due to increased flow during EPU conditions. Loads for Main Transformer controls/cooling equipment and Isolated-Phase Bus cooling equipment increase due to increased cooling requirements. Revised electrical system calculations were required to address the load increases. Based on these revised calculations, the existing load shedding scheme [actuated upon a Loss-of-Coolant Accident (LOCA) event] is expanded to actuate during generator trip events (non-LOCA). This provides additional protection against inadequate voltages on the emergency buses during potential degraded grid events. Administrative load management in conjunction with the additional load shedding results in increased voltage levels for design basis events. Under normal conditions, administrative load management ensures that the ratings of electrical supply and distribution components (switchgear, motor control centers, cables, etc.) are adequate. In addition, there is a minimal effect on short circuit current values and all values continue to be acceptable for EPU conditions.

Station loads under emergency operation/distribution conditions (emergency diesel generators) are based on equipment nameplate data, except for the Emergency Core Cooling System (ECCS) pumps where a conservatively high flow brake horsepower (BHP) is used. Operation at the EPU RTP level is achieved by utilizing existing equipment operating at or below the nameplate rating and within the calculated BHP for the stated pumps; therefore, under emergency conditions the electrical supply and distribution components are adequate.

No increase in flow or pressure is required of any AC-powered ECCS equipment for the EPU. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) is not increased with the EPU, and the current emergency power system remains adequate. The systems have sufficient capacity to support all required loads for safe shutdown, to maintain a safe shutdown condition, and to operate the engineered safety feature equipment following postulated accidents.

6.2 Direct Current Power

Operation at the EPU RTP level does not increase any loads beyond nameplate rating or design basis loading, nor revise any control logic; therefore the DC power distribution system is adequate.

6.3 Fuel Pool

The EPU does not affect the heat removal capability of the Fuel Pool Cooling and Cleanup (FPCC) system. The EPU results in slightly higher core decay heat loads during refueling. The higher decay heat loads could result in a slight delay in removing RHR system from service.

The EPU analysis assumes a 24-month fuel cycle length and GE14 fuel as the basis. Each reload affects the decay heat generation in the SFP after discharge of fuel from the reactor. This evaluation considered the expected heat load in the SFP at the EPU conditions, and confirms the capability of the FPCC to maintain adequate fuel pool cooling.

The normal radiation levels around the SFP may increase slightly, primarily during fuel handling operations. This increase is acceptable and does not significantly increase the operational doses to personnel or equipment. There is no effect on the design of the spent fuel racks, because the SFP design temperature is not exceeded.

6.4 Water Systems

Evaluations of the service water systems were performed to determine the effect of the EPU on these systems. The results of these evaluations concluded that the safety-related and nonsafety-related service water system capabilities are adequate, and the environmental effects of EPU are controlled at the current level. This conclusion is based on the following considerations.

The safety-related service water systems are designed to provide a reliable supply of cooling water during and following a DBA for the following essential equipment and systems:

RHR heat exchangers;

Emergency diesel-generator coolers;

Cooling units for the CS and RHR pump rooms.

Evaluations show that the implementation of the EPU does not require a change to the safety-related service water systems.

Regarding the nonsafety-related heat loads, the heat rejected to the Service Water system via the closed cooling water systems and other auxiliary heat loads increases from the EPU due to an increase in main generator losses rejected to the stator water coolers and increased bus cooler heat loads. These additional heat loads are a minor portion of the total service water system heat load and result in a negligible discharge temperature increase.

For normal operation, the maximum service water and circulating water heat loads occur during peak summer months. An EPU discharge temperature was estimated assuming both realistic conditions and very conservative bounding conditions. The results demonstrate that the service water system and circulating water system are adequate for the EPU conditions.

Performance of the main condenser was evaluated for EPU. This evaluation was based on a design duty over the actual yearly range of circulating water inlet temperatures, and confirms that the condenser and circulating water system are adequate for EPU operation.

The heat loads on the Reactor Building Closed Cooling Water (RBCCW) system are not increased significantly by the EPU because they depend mainly on either vessel temperature or flow rates in the systems cooled by the RBCCW. The change in vessel temperature is minimal and does not result in any significant increase in drywell cooling loads. The Recirculation and RWCU pump drive flow rates do not significantly change, and thus, the pump cooling needs are effectively unchanged by the EPU.

The heat loads on the Turbine Building Closed Cooling Water (TBCCW) system that are power-dependent and are increased by the EPU are those related to the operation of the turbine-generator. The remaining TBCCW heat loads are not strongly dependent upon reactor power and do not increase significantly. The TBCCW contains sufficient redundancy to assure that adequate heat removal capability is always available. Therefore, sufficient cooling capacity for EPU operation is available.

The Ultimate Heat Sink (UHS) for intake is the Cape Fear River estuary and discharge is the Atlantic Ocean. As a result of operation at the EPU RTP level, the post-LOCA UHS water temperature does not increase.

A review was performed to evaluate the increased UHS heat load for the EPU. The review concludes that the existing UHS system provides a sufficient quantity of water at a temperature less than 92°F (design temperature) following a design basis LOCA. The current Technical Specifications for UHS limits are adequate, due to conservatism in the original design.

The state thermal discharge limits were compared to the current discharges and bounding analysis discharges for the EPU. As a result, the National Pollutant Discharge Elimination System (NPDES) Permit limit regarding plume area temperature measured in the Atlantic Ocean will be revised to increase the thermal plume mixing zone acreage and extents. With this change, the plant will remain within the state discharge limit during EPU operation.

6.5 Standby Liquid Control System

The operating capability of the Standby Liquid Control System (SLCS) is unaffected by the EPU. However, a new fuel design combined with the expected fuel cycle operating time requires an increase in the minimum reactor boron concentration from 660 ppm to 720 ppm after the first EPU operating cycle. Associated Technical Specification changes will be addressed in a separate licensing amendment from the EPU.

The increase in the reactor boron concentration requirement from 660 ppm to 720 ppm for subsequent cycles necessitates changes to the storage parameters for the neutron absorber solution. The neutron absorber injection rate requirement for maintaining the peak suppression pool water temperature limits following the limiting ATWS event with SLCS injection is not increased. In addition, the solution concentration level in the storage tank is being reduced to lessen the dependence on the tank heaters and the line trace heaters, and to change the method of compliance with 10 CFR 50.62 to eliminate the requirement to operate both pumps simultaneously. The changes to the solution concentration, solution volume available for injection, the Boron-10 enrichment, and the number of pumps required to be in operation, are being implemented coincident with the EPU.

Implementation of the EPU has no adverse effect on the ability of the SLCS to mitigate an ATWS.

6.6 Power-Dependent Heating Ventilation and Air Conditioning

The HVAC systems consist mainly of heating, cooling supply, exhaust and recirculation units in the reactor building, drywell, and turbine building. EPU operation is expected to result in slightly higher process temperatures and a small increase in the heat load due to higher electrical current in some motors and cables.

The areas most affected due to the increase in process temperatures from extraction steam, condensate, feedwater, and/or motor horsepower are: The 1A and 1B Feedwater Heater and condenser area in the Turbine Building and the areas immediately surrounding the condensate and condensate booster pump motors. Other areas are minimally affected (< 2°F) by the EPU because the process temperatures remain relatively constant.

Heat loads in the drywell increase slightly due to increases in the recirculation pump motor horsepower and the feedwater process temperature. The maximum temperature increase in the drywell is 1.8°F.

The heat loads discussed above represent an increase of approximately 2% to 5% in the drywell cooling, main steam isolation valve (MSIV) valve pit, radwaste building, and main steam line tunnel and approximately 14% in the feedwater heater area heat loads. Based on a review of design basis calculations and environmental qualification design temperatures, the above increases are within the excess design capability available. Therefore, the design and operation of the HVAC is not adversely affected by the EPU.

6.7 Fire Protection

Operation of the plant at the EPU RTP level does not affect the fire suppression or detection systems. Any changes in physical plant configuration or combustible loading as a result of modifications to implement the EPU, will be evaluated in accordance with plant modification and fire protection programs. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not require modification, and are adequate for the EPU conditions. One of the required evaluations assumed that operators would increase the RCIC flow controller setpoint to 500 gpm. With the SRVs assumed to be regulating vessel pressure based on nominal setpoints, the existing RCIC configuration can provide this increased flow. Other than increasing the RCIC flow, operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire protection systems and analyses are not affected by the EPU.

A plant-specific evaluation was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R assuming EPU conditions. The results of the Appendix R evaluation for the EPU demonstrate that fuel cladding integrity, RPV integrity and containment integrity are maintained, and that sufficient time is available for the operator to perform the necessary actions. No changes are required in the equipment required for safe shutdown for the Appendix R event. Therefore, the EPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of an Appendix R fire event, and satisfies the requirements of Appendix R with respect to achieving and maintaining safe shutdown in the event of a fire.

6.8 Systems Not Affected By EPU

The following systems are not affected by operation of the plant at the EPU condition:

1. Reactor Protection System[1]
2. Post Accident Sampling System

[1] The Reactor Protection System (RPS) was originally identified in ELTR1, Table J-3 as a system dependent on power level. Current EPUs now classify the RPS system as not affected, because the system logic is unchanged. Changes to RPS trip setpoints occur within the originating systems (e.g., Neutron Monitoring System).

3. Torus Drain System
4. Auxiliary Boiler System
5. Turbine Building Sampling System
6. Screen Wash System
7. Emergency AC Lighting System
8. Annunciator and Remote Annunciator Systems
9. Caswell Beach Supervisory and Control System
10. Service Air System
11. Pneumatic Nitrogen System
12. Hydrogen Supply System
13. Carbon Dioxide Supply System
14. Lube Oil Storage and Transfer System
15. Potable Water System
16. Radwaste Sampling System
17. Refueling System
18. Reactor Vessel Service Equipment, New Fuel Storage
19. Spent Fuel System
20. Grounds Maintenance/Landscaping
21. Clean Machine Shop
22. Service Water Building
23. HVAC Service Building
24. Augmented Off Gas Building
25. Auxiliary Boiler House
26. Diesel Generator Building
27. Control Building
28. Radwaste Building
29. Water Treatment Building
30. Miscellaneous Structures or Out Buildings

31. Safety Equipment
32. General Instrumentation and Control Spares
33. General Mechanical Spares

Some BSEP systems are affected to a very small extent by operation of the plant at the EPU condition. For these systems, the effects are insignificant to the design or operation of the system and equipment:

1. Reactor Building Sampling
2. Condensate Piping and Valves and Condensate Return System
3. Condensate Makeup System
4. Diesel Generator Fuel Oil System
5. Site Cables (Wiring, Trays, and Conduit)
6. Main Control Board
7. Auxiliary Control Board
8. Instrument Air System
9. Fuel Oil System
10. Water Treatment System
11. Demineralized Water System
12. Hydrogen Water Chemistry
13. Reactor Building
14. Turbine Building

Table 6-1
EPU Plant Electrical Characteristics

Parameter	Value
Gross Generator Output (MWe)	1,006.4
Rated Voltage (KV)	24
Power Factor	0.95
Generator Rated Output (MVA)	1,039
Current Output (A)	25,089 (BSEP 1) / 25,311 (BSEP 2)
Isolated Phase Bus Duct Rating (A):	
Main Section	>25,311(force-cooled) / 13,750 (self-cooled)
Branch Section	14,613 (force-cooled) / 7,948 (self-cooled)
Main Transformers Rating (MVA)	400 (individual) / 1,200 (bank)
EPU Transformer Output – Bank (MVA)	< 1,039

7 POWER CONVERSION SYSTEMS

The power conversion systems were originally designed to utilize the energy available from the NSSS and were designed to accept the system and equipment flows resulting from continuous operation at 105% of rated steam flow. However, the structural capabilities of the power conversion systems allow for steam flows greater than 105% of original rated steam flow.

7.1 Turbine-Generator

The turbine and generator were originally designed with a maximum flow-passing capability and generator output in excess of rated conditions to ensure that the original rated steam-passing capability and generator output is achieved. This excess design capacity ensures that the turbine and generator meet rated conditions for continuous operating capability with allowances for variations in flow coefficients from expected values, manufacturing tolerances, and other variables that may adversely affect the flow-passing capability of the units. The difference in the steam-passing capability between the design condition and the rated condition is called the flow margin.

The turbine-generator was originally designed with a flow margin of 5%, and modifications will be made as needed to the turbine to maintain a minimum flow margin of 3%. Besides the modification to the high pressure turbine, only minor (non-safety) modifications are needed.

Both the low and high pressure turbine sections are integral or monoblock rotors which are not considered a source for potential missile generation, and therefore, neither a high pressure nor a low pressure turbine rotor missile probability analysis is required.

7.2 Condenser And Steam Jet Air Ejectors

The performance of the main condensers was evaluated for EPU with the following conclusions.

- Both condenser hotwell capacities and level instrumentation are adequate for the EPU condition.
- The condensers are considered adequately protected against tube vibration damage at the EPU conditions.
- The design of the condenser air removal system is not adversely affected and no modification to the system is required. The design capacity of the steam jet air ejectors

(SJAEs) is not affected because they were originally designed for operation at significantly greater than warranted flows.

7.3 Turbine Steam Bypass

For BSEP 1, the turbine bypass system was rated for a total steam flow capacity of approximately 23.79% of the current RTP reactor steam flow, or 2.638 Mlb/hr. At EPU RTP, rated reactor steam flow is 12.781 Mlb/hr, resulting in a bypass capacity of 20.6%.

For BSEP 2, the turbine bypass system was rated for a total steam flow capacity of approximately 80.26% of the current RTP reactor steam flow, or 8.9 Mlb/hr, resulting in a bypass capacity of 69.6% of EPU rated steam flow. The BSEP 2 Bypass system is still considered operable with two out of the ten bypass valves OOS. The BSEP 2 Bypass system is capable of accepting approximately 55.5% of the EPU rated steam flow with two bypass valves OOS.

The steam bypass system is a nonsafety-related system. Even though the bypass capacity as a function of the percent uprated steam flow is reduced, the actual steam bypass capacity is unchanged. This capacity is used in the transient analysis (Section 9.1) for the evaluation of events that credit the turbine bypass system availability. Because the EPU transient analysis results are acceptable, the turbine bypass capacity is adequate for EPU operation.

7.4 Feedwater And Condensate Systems

The feedwater and condensate systems do not perform a system level safety-related function, and are designed to provide a reliable supply of feedwater at the temperature, pressure, quality, and flow rate as required by the reactor. Therefore, these systems are not safety-related. Their performance does, however, have a major effect on plant availability and capability to operate at the EPU condition.

Some modifications to nonsafety-related equipment in the feedwater and condensate systems are necessary to attain full EPU core thermal power. The changes in operating pump line-ups and equipment modifications may include placing additional pumps in operation, resizing the feedwater and condensate pump impellers, pump motors, and motor/pump couplings, and/or possible modifications of the feedwater turbines. Furthermore, there may be modifications to some nonsafety-related equipment in the Onsite Electrical Distribution system to accommodate additional loads during normal and off-normal conditions. Implementation of these modifications will be reviewed under the 10 CFR 50.59 process. In addition, a review of these

modifications will be conducted to confirm that they do not constitute a material alteration, as discussed in 10 CFR 50.92.

During steady-state conditions, the feedwater and condensate systems will have adequate NPSH for all of the pumps to operate without cavitation in the EPU conditions.

The feedwater heaters were evaluated to confirm that pressures and temperatures at the EPU conditions do not exceed the original equipment design values and therefore are expected to remain within their design envelope for thermal stress and strain. The performance of the feedwater heaters will be monitored for indications of unacceptable vibration response during the EPU power ascension program.

The effect of the EPU on the condensate polishing demineralizers (CPDs) was reviewed. In summary, the system is adequate for EPU operation, but will experience slightly higher loadings. The higher loadings result in reduced CPD run times. However, the reduced run times are acceptable.

8 RADWASTE SYSTEMS AND RADIATION SOURCES

8.1 Liquid Waste Management

Based on a review of plant operating effluent reports and the slight increase expected from EPU, it is concluded that the requirements of 10 CFR 20 and 10 CFR 50, Appendix I are expected to be met. Therefore, the EPU does not have an adverse effect on the processing of liquid radwaste, and there are no significant environmental effects.

8.2 Gaseous Waste Management

The Gaseous Waste Management systems collect, control, process, store, and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. The gaseous waste management systems include the Offgas system and various building ventilation systems. The systems are designed to meet the requirements of 10 CFR 20 and 10 CFR 50, Appendix I.

The non-condensable gases (which primarily consist of N-13, N-16, O-19 and various noble gases) are continuously removed from the main condensers by the SJAEs, which discharge into the Offgas system.

The activity of airborne effluents released through building vents is not expected to increase significantly with the EPU. The release limit is an administratively controlled variable, and is not a function of core power. The gaseous effluents are well within limits at original power operation and remain well within limits following implementation of the EPU.

Core radiolysis (i.e., formation of H_2 and O_2) increases linearly with core power, thus increasing the heat load on the recombiner and related components. Based on a heat balance for the offgas recombiner under current rated power conditions, the radiolytic hydrogen flow rate increases, but remains well within the design capacity of the system.

8.3 Radiation Sources in the Reactor Core

During power operation, the radiation sources in the core include radiation from the fission process, accumulated fission products and neutron reactions as a secondary result of fission. Historically, these sources have been defined in terms of energy released per unit of reactor power. Therefore, the increase in the operating source term is proportional to the increase in power.

For post-operation evaluations, two sets of source data are applied. The first is the gamma-ray source, which is used in shielding calculations for the core and for individual fuel bundles. This set of source terms increases in proportion to reactor power. The second is used for post-accident evaluations, which are performed in compliance with regulatory guidance that applies different release and transport assumptions to different fission products. Plant-specific fission product inventories were developed and used in the evaluation of design basis accidents.

8.4 Radiation Sources in the Reactor Coolant

The reactor coolant contains activated corrosion products, which are the result of metallic materials entering the water and being activated in the reactor region. In addition, BSEP 1 and 2 are pumped forward plants in which only about 67% of total steam flow passes through the condensate demineralizer. Under the EPU conditions, the feedwater flow increases with power, the activation rate in the reactor region increases with power, and the filter efficiency of the condensate demineralizers may decrease as a result of the feedwater flow increase. The net result may be an increase in the activated corrosion product production. However, the design basis for BSEP 1 and 2 includes the tables from NUREG-0016 (Reference 4). The Offsite Dose Calculation Manual (ODCM) calculations include the adjustment factors required for application to BSEP. Therefore, no change is required in the design basis activated corrosion product concentrations for the EPU.

8.5 Radiation Levels

For the EPU, normal operation radiation levels increase slightly. For conservatism, many aspects of the plant were originally designed for higher-than-expected radiation sources. Thus, the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant, because it is offset by conservatism in the original design basis source terms used and analytical techniques.

Post-operation radiation levels in most areas of the plant are expected to increase by no more than the percentage increase in power level. In a few areas, the increase could be slightly higher. Individual worker exposures should be maintained within acceptable limits by the site ALARA program, which controls access to radiation areas. Procedural controls compensate for increased radiation levels. In addition, the plant has established successful zinc injection and Hydrogen Water Chemistry (HWC) programs, which result in a decrease in post-operation radiation levels and/or reduced repairs required in radiation areas.

The change in core inventory resulting from the EPU and application of the AST increases the plant's design basis post-accident radiation levels by a factor approximately equal to the power

level increase. A review of areas requiring post-accident occupancy (per NUREG-0737 Item II.B) concluded that access needed for accident mitigation is not significantly affected by the EPU. The post-accident habitability of the Emergency Operations Facility / Technical Support Center was also evaluated and demonstrated to remain within regulatory dose limits.

For the EPU, normal operation gaseous activity levels increase slightly. The increase in activity levels is generally proportional to the percentage increase in core thermal power. The Technical Specification limits implement the guidelines of 10 CFR 50, Appendix I. A review of the normal radiological effluent doses shows that at original power, the doses are a small fraction of the doses allowed by Technical Specification limits. The EPU does not involve significant increases in the offsite dose from noble gases, airborne particulates, iodine, tritium or liquid effluents. In addition, radiation from shine is not a significant exposure pathway. Present offsite radiation levels are a negligible portion of background radiation. Therefore, the normal offsite doses are not significantly affected by operation at the EPU RTP level and remain below the limits of 10 CFR 20 and 10 CFR 50, Appendix I.

9 REACTOR SAFETY PERFORMANCE EVALUATIONS

9.1 Reactor Transients

The UFSAR evaluates the effects of a wide range of potential plant transients. Disturbances to the plant caused by a malfunction, a single equipment failure, or an operator error are evaluated according to the type of initiating event per Regulatory Guide 1.70, Chapter 15.

Most of the transient events are analyzed at the full EPU RTP and maximum allowed core flow operating point on the power-flow map. Analytical results demonstrate the capability of the design to meet all transient safety criteria for EPU RTP conditions.

The cycle-specific OLMCPRs will be supplied in the Core Operating Limit Reports (COLRs). The historical 25% of RTP value for the Technical Specification Safety Limit, some thermal limits monitoring Limiting Conditions for Operation (LCOs) thresholds, and some Surveillance Requirements (SRs) thresholds is based on generic analyses (evaluated up to ~50% of original RTP) applicable to the plant with highest average bundle power (the BWR6) for all of the BWR product lines. As a result of the EPU, the Safety Limit percent RTP basis, some thermal limits monitoring LCOs, and some SR percent RTP thresholds are reduced to 23% RTP.

The LOFW transient, assuming an additional single failure (loss of HPCI), was analyzed for the EPU. During this low probability event, reactor water level is automatically maintained above the TAF by the RCIC system, without credit for any other injection system. Operator action is needed to inhibit the ADS actuation. After water level is restored, the operator would manually control water level, reduce reactor pressure, and initiate RHR shutdown cooling.

9.2 Design Basis Accidents

The radiological consequences of the plant design basis LOCA have been reviewed for the effect of the EPU using the guidance of RG 1.183 AST as described in Reference 5. The results are within the guidelines of 10 CFR 50.67 and GDC 19 of 10 CFR 50, Appendix A. Other accidents (non-LOCA) analyzed in the UFSAR have also been reviewed and remain below their regulatory limits. NRC review of the implementation of the AST is being addressed in a separate licensing amendment from the EPU.

The events analyzed were the LOCA, the Main Steam Line Break Accident (MSLBA) outside containment, the Fuel Handling Accident (FHA), and the Control Rod Drop Accident (CRDA). The plant-specific results for the EPU are shown in Tables 9-1 through 9-4.

9.3 Special Events

A BSEP-specific ATWS analysis for the EPU condition was performed resulting in the peak vessel pressure, peak clad temperature, peak clad oxidation, peak suppression pool temperature, and peak containment pressure meeting the acceptance criteria. Therefore, the plant response to an ATWS event during EPU operation is acceptable.

Plant response to and coping capabilities for a station blackout (SBO) event are affected slightly by operation at the EPU RTP level, due to an increase in the decay heat. There are no changes to the systems or equipment used to respond to an SBO, nor is the required coping time changed. The plant continues to meet the requirements of 10 CFR 50.63 after the EPU.

Table 9-1
LOCA Radiological Consequences

Location	**EPU / AST**	**Limit**
Exclusion Area: Dose (rem TEDE)	0.61	≤ 25
Low Population Zone: Dose (rem TEDE)	1.34	≤ 25
Control Room: Dose (rem TEDE)	3.40	≤ 5

Table 9-2
MSLBA Radiological Consequences

Location	**EPU / AST**	**Limit**
Case 1: Iodine concentration in coolant = 4 μCi/gm dose-equivalent I-131		
Exclusion Area: Dose (rem TEDE)	2.52	≤ 25
Low Population Zone: Dose (rem TEDE)	0.89	≤ 25
Control Room: Dose (rem TEDE)	0.50	≤ 5
Case 2: Iodine concentration in coolant = 0.2 μCi/gm dose-equivalent I-131		
Exclusion Area: Dose (rem TEDE)	0.127	≤ 2.5
Low Population Zone: Dose (rem TEDE)	0.045	≤ 2.5
Control Room: Dose (rem TEDE)	0.025	≤ 5

Table 9-3
FHA Radiological Consequences

Location	**EPU / AST**	**Limit**
Fuel Handling Accident (Single fuel bundle dropped)		
Exclusion Area: Dose (rem TEDE)	5.51	≤ 6.25
Low Population Zone: Dose (rem TEDE)	1.95	≤ 6.25
Control Room: Dose (rem TEDE)	2.69	≤ 5

Table 9-4
CRDA Radiological Consequences

Location	**EPU / AST**	**Limit**
Exclusion Area: Dose (rem TEDE)	0.27	≤ 6.25
Low Population Zone: Dose (rem TEDE)	0.22	≤ 6.25
Control Room: Dose (rem TEDE)	0.28	≤ 5

10 ADDITIONAL ASPECTS OF EXTENDED POWER UPRATE

10.1 High Energy Line Break

The evaluation of the piping systems defined in the UFSAR as high energy systems determined that there is no change in postulated break locations. No mass and energy release rate increases result from the postulated high energy line breaks (HELBs) with steam. Operation at the EPU RTP level requires an increase in the steam and feedwater flows, which results in a slight increase in downcomer subcooling. This, in turn, results in a small increase in the mass and energy release rates following HELBs starting with subcooled liquid.

The evaluation shows that the affected building and cubicles that support a safety-related function are designed to withstand the resulting pressure and thermal loading following an HELB. Therefore, the changes in HELBs due to the EPU do not affect the environmental qualification of equipment and systems that support a safety-related function.

Existing calculations for the development of pipe whip and jet impingement loads from the postulated HELBs have been determined to be bounding for the safe shutdown of the plant in the EPU condition. Therefore, existing pipe whip restraints and jet impingement shields, and their supporting structures are adequate for the EPU conditions.

There is no effect on the plant internal flooding analysis or safe shutdown analysis due to EPU.

A Moderate Energy Line Break (MELB) break analysis is not within the BSEP licensing basis. Therefore, MELB is not applicable to BSEP 1 and 2 for the EPU.

10.2 Equipment Qualification

The safety-related electrical equipment was reviewed to assure that the existing qualification for the normal and accident conditions expected in the area where the devices are located remains adequate.

The implementation of the EPU includes the use of an AST based on RG 1.183 as the BSEP 1 and 2 design and licensing basis for evaluating offsite and control room doses. However, the evaluation of the effect of the EPU on Equipment Qualification (EQ) is based on the interim guidance given in SECY-99-240 where the continued use of the TID-14844 post-accident source term release is considered acceptable for evaluating proposed plant modifications on previously-analyzed integrated component doses, regardless of the accident source term used to evaluate offsite and control room doses.

EQ for safety-related electrical equipment located inside the containment is based on MSLBA and/or DBA LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environment expected to exist during normal plant operation. The current accident conditions for temperature and pressure are modified for the EPU RTP conditions. Normal temperatures are expected to increase slightly and will be evaluated through the EQ temperature monitoring program. Current radiation levels under normal plant conditions were conservatively evaluated to increase approximately 20%. Radiation levels under accident conditions were conservatively evaluated to increase 15%.

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from a main steam line break in the pipe tunnel, or other high energy line breaks, whichever is limiting for each plant area. The accident temperature, pressure and humidity conditions resulting from a LOCA or HELB do not change for the EPU. The accident temperature and pressure levels do not increase due to operation at the EPU RTP level. The normal temperature does not change significantly. The normal pressure, humidity, and radiation levels do not change as a result of the EPU.

The EQ review for the EPU conditions identified some equipment located inside and outside the containment, which could potentially be affected by the higher radiation levels. The qualification of this equipment will be resolved by reanalysis, by refined radiation calculations (location specific), by slightly reducing qualified life, by adding new equipment, or by replacing the existing equipment with qualified equipment.

10.3 Mechanical Component Design Qualification

Operation at the EPU RTP level increases the normal ambient temperature less than 2°F. The accident radiation level and the normal radiation level also increase due to the EPU. Safety-related mechanical equipment with non-metallic components were reevaluated and were determined to be acceptable for the environmental conditions associated with the EPU.

The mechanical design of equipment/components (e.g., pumps, heat exchangers) in certain systems is affected by operation at the EPU RTP level due to slightly increased temperatures and flows. The revised operating conditions do not significantly affect the cumulative usage fatigue factors of mechanical components. Therefore, the mechanical components and component supports are adequately designed for the EPU conditions.

10.4 Required Testing

The following testing will be performed at the time of implementation of EPU:

- Surveillance testing will be performed on the instrumentation that requires re-calibration for the EPU in addition to the testing performed according to the plant Technical Specifications schedule.

- During the power ascension in which the current RTP will be exceeded, steady-state data will be taken starting from 90% of the current RTP up to the EPU RTP, so that system performance parameters can be projected throughout the EPU power ascension.

- Power increases beyond the previous RTP will be made in increments of ≤ 5%. Steady-state operating data will be taken and evaluated at each step.

- Control system checks will be performed for the feedwater/reactor water level controls and pressure controls. These operational tests will be made at the appropriate plant conditions for that test and at each power increment above the previous rated power condition, to show acceptable operational capability.

The same performance criteria shall be used as in the original power ascension tests, except where updated to be consistent with the modified plant configuration.

BSEP does not intend to perform testing regarding initiating an automatic scram from high power. The operating history of the plants has shown previous transient events from full power to be within expected peak limiting values. The transient analysis performed for the EPU demonstrates that all safety criteria are met and that the EPU does not cause any previous non-limiting events to become limiting. Performing such testing given the available information is considered non-conservative and an unnecessary challenge to reactor safety systems. If any large transient were to occur, plant procedures require verification that the plant responded in accordance with expected responses with respect to the UFSAR. Existing plant event data recorders are capable of capturing plant data to confirm expected responses.

The piping vibration levels of the main steam system piping and the feedwater system piping in each plant will be monitored during initial plant operation at the new EPU operating conditions. These piping systems will be monitored for vibration because the mass flow rates in these piping systems will increase during EPU operations. The mass flow rates in these systems will increase approximately in proportion to the power level increase. The startup vibration test program will

show that these piping systems are operating at acceptable vibration stress levels during initial plant operation at the EPU conditions.

The plant 10 CFR 50 Appendix J test program is required by the Technical Specifications and is described in UFSAR Section 6.2. This test program periodically pressurizes the containment (Type A test), the containment penetrations (Type B test), and the containment isolation valves and test boundary (Type C tests) to the calculated peak containment pressure (P_a), and measures leakage. The current value of P_a is 49 psig, which bounds the calculated peak containment pressure of 46.4 psig for EPU. Therefore, the current P_a value of 49 psig for the 10 CFR 50 Appendix J test program remains unchanged.

10.5 Individual Plant Evaluation

BSEP developed Level 1 and Level 2 Probabilistic Safety Assessment (PSA) models and submitted the analyses to NRC as the Individual Plant Examination (IPE) and Individual Plant Examination – External Events (IPEEE) Submittals. BSEP has maintained these PSA models to conform to plant configuration and operating procedure changes subsequent to the original development, i.e., it is a "living PSA."

Changes due to EPU implementation were evaluated for the effect on the PSA models in the following key areas:

- Initiating Event Frequency
- Component Reliability
- Success Criteria
- Operator Response

The evaluation concluded that EPU implementation does not change initiating event frequencies or component reliability assumed in the current PSA. Further, while some plant parameters are affected by EPU implementation, these changes were within the existing margin of the current success criteria in the PSA and revisions were not required in order to satisfy the overall safety success criteria. Finally, operator response time was slightly changed for some events. The effect of the EPU on the plant risk profile was insignificant.

10.6 Operator Training And Human Factors

Additional training required to operate the plant in the EPU condition is expected. The changes to the plant have been identified and the operator training program is being evaluated to

determine the specific changes required for operator training. This evaluation includes the effect on the plant simulator.

For EPU RTP conditions, operator responses to transient, accident and special events are not affected. The EPU does not change any of the automatic safety functions. After the applicable automatic responses have initiated, the follow on operator actions (e.g., maintaining safe shutdown, core cooling, and containment cooling) for plant safety do not change for the EPU.

Training required to operate the plant following the EPU will be conducted prior to restart of the unit at the EPU conditions. Data obtained during startup testing will be incorporated into additional training as needed. The classroom training will cover various aspects of the EPU including changes to parameters, setpoints, scales, procedures, systems, and startup test procedures. The classroom training will be combined with simulator training as appropriate. The simulator training as a minimum will include a demonstration of transients that show the greatest change in plant response at the EPU RTP compared to current power. Simulator changes and fidelity revalidation will be performed in accordance with ANSI/ANS 3.5-1998.

10.7 Plant Life

The longevity of most equipment is not affected by the EPU. There are various plant programs (EQ, FAC) that deal with age-related components. These programs were reviewed, and do not significantly change for the EPU. In addition, the Maintenance Rule provides oversight for the other mechanical and electrical components, important to plant safety, to guard against age-related degradation.

11 LICENSING EVALUATIONS

11.1 Evaluation Of Other Applicable Licensing Requirements

The analysis, design, and implementation of EPU were reviewed for compliance with the current plant licensing basis acceptance criteria and for compliance with new regulatory requirements and operating experience in the nuclear industry. Plant unique evaluations have been performed for the subjects addressed below.

All of the issues raised by the following sources were evaluated on a plant-specific basis as part of the EPU program. These evaluations conclude that every issue is either: (1) not affected by the EPU, (2) already incorporated into the generic EPU program, or (3) bounded by the plant-specific EPU evaluations.

Code of Federal Regulations (CFR)

NRC TMI Action Items

NRC Action Items (Formerly Unresolved Safety Issues) and New Generic Issues

NRC Regulatory Guides

NRC Generic Letters

NRC Bulletins

NRC Information Notices

NRC Circulars

INPO Significant Operating Experience Reports (applicable to the EPU)

GE Services Information Letters

GE Rapid Information Communication Service Information Letters

Plant-unique items whose previous evaluations could be affected by operation at the EPU RTP level are reviewed in accordance with existing BSEP design control processes. These items include (1) the NRC and Industry communications discussed above, (2) the safety evaluations for work in progress and not yet integrated into the plant design, (3) the temporary modifications that were in place prior to the EPU and will remain in place after EPU implementation, and (4) the plant emergency operating procedures (EOPs). These items are reviewed for possible effect of the EPU and are revised to reflect the EPU conditions, as applicable.

11.2 Affect On Technical Specifications

Implementation of the EPU requires revision of a number of the Technical Specifications. Table 11-1 contains a list of Technical Specification items, which are changed to implement the EPU.

11.3 Environmental Assessment

An assessment of environmental impacts of the BSEP EPU has been performed. This assessment compared the environmental impacts of the EPU to those previously identified by the U. S. Atomic Energy Commission in the (1974) Final Environmental Statement (FES) for continued construction and proposed issuance of an operating license for BSEP and the (1997) Environmental Assessment for a 5 percent thermal power uprate.

The non-radiological environmental effects resulting from the BSEP EPU are minimal. The EPU will be implemented without making extensive changes to plant systems that directly or indirectly interface with the environment. None of the necessary modifications will involve land disturbance or new construction outside of established facility areas. There will be no change in the amount of water withdrawn from the Cape Fear River for condenser cooling, and only a relatively small increase in the amount of waste heat discharged to the Atlantic Ocean. CP&L has submitted an application, to the State of North Carolina, for a revision to its NPDES permit. This revision, in part, adjusts the established ocean discharge mixing zone to account for the slight increase in circulating water temperatures. Thermal limits established in the NPDES permit are not revised. The overall effect of this discharge temperature increase on ocean temperature is negligible.

There are no significant radiological environmental effects resulting from the BSEP EPU. The radioactive waste systems at BSEP are designed to collect, process, and dispose of radioactive wastes in a controlled and safe manner. The design bases for these systems during normal operation are to limit discharges in accordance with 10 CFR 20, to limit exposures to the requirements of 40 CFR 190, and to satisfy the design objectives of 10 CFR 50 Appendix I. Adherence to these limits and objectives continue after EPU. Operation at EPU conditions does not result in any physical changes to the solid waste, liquid waste, or gaseous waste systems. The safety and reliability of these systems are unaffected by the EPU. Also, EPU does not affect the environmental monitoring of any of these waste streams or the radiological monitoring requirements of the BSEP Technical Specifications. EPU does not introduce any new or different radiological release pathways and does not increase the probability of an operator error

or equipment malfunction that would result in an uncontrolled radioactive release from the radioactive waste streams.

No significant change in the volume or activity of water treated and released is expected as a result of EPU. It is expected that gaseous effluents will increase slightly as a result of EPU. BSEP offsite doses for the previous five years of operation have been well below the 10 CFR 50 Appendix I standards and remain well below these limits after EPU. All offsite radiation doses are small and within applicable regulatory standards.

11.4 Significant Hazards Consideration Assessment

11.4.1 Introduction

Uprating the power level of nuclear power plants can be done safely within certain plant-specific limits, and is an extremely cost effective way to increase the installed electricity generating capacity. Several light water reactors have already been uprated world wide, including numerous boiling water reactors (BWRs) in the United States, Switzerland and Spain.

The significant safety analyses and evaluations have been performed, and their results justify uprating the licensed thermal power at BSEP 1 and 2 by 14.3% to 2923 MWt.

11.4.1.1 Modification Summary

An increase in electrical output of a BWR plant is accomplished primarily by generation and supply of higher steam flow for the turbine generator. Most BWR plants, as currently licensed, have an as-designed equipment and system capability to accommodate steam flow rates at least 5% above the original rating. In addition, continuing improvements in the analytical techniques and computer codes based on several decades of BWR safety technology, plant performance feedback, operating experience, and improved fuel and core designs have resulted in a significant increase in the design and operating margins between calculated safety analysis results and the licensing limits. These available safety analysis differences, combined with the excess as-designed equipment, system and component capabilities, provide BWR plants with the capability to increase their thermal power ratings between 5 and 10% without major Nuclear Steam Supply System (NSSS) hardware modifications, and to provide for power increases to 20% with limited non-safety hardware modifications, with no significant increase in the hazards presented by the plant as approved by the Nuclear Regulatory Commission (NRC) at the original license stage.

The plan for achieving higher power is to expand the operating envelope on the power/flow map through extension of the existing implementation of Maximum Extended Load Line Limit Analysis (MELLLA). However, there is no increase in the maximum core flow limit or operating pressure over the pre-extended power uprate (EPU) values. For EPU operation, the plant already can readily be modified to have adequate control over inlet pressure conditions at the turbine, to account for the larger pressure drop through the steam lines at higher flow and to provide sufficient pressure control and turbine flow capability.

11.4.2 Discussions of Issues Being Evaluated

Plant performance and responses to hypothetical accidents and transients have been evaluated for an EPU license amendment. This safety assessment summarizes the safety significant plant reactions to events analyzed for licensing the plant, and the potential effects on various margins of safety, and thereby concludes that no significant hazards consideration will be involved.

11.4.2.1 EPU Analysis Basis

BSEP 1 and 2 were originally licensed at 2436 MWt. The plants are currently licensed for a 100% Rated Thermal Power (RTP) level of 2558 MWt [105% of the Original Licensed Thermal Power (OLTP)]. The current accident analyses were generally performed at 102% of the current RTP level, in accordance with Regulatory Guide (RG) 1.49. Some analyses were performed at 100% EPU RTP, because the 2% power factor of RG 1.49 is already accounted for in the analysis methodology. The EPU RTP level (2923 MWt) included in this evaluation is a 14.3% thermal power increase from current RTP. Similar to current analyses, the EPU safety analyses are based on a power level of at least 1.02 times the EPU RTP level, except where it is already accounted for in the analysis methodology.

11.4.2.2 Margins

The above EPU analysis basis ensures that the power dependent margins prescribed by the Code of Federal Regulations (CFRs) are maintained by meeting the appropriate regulatory criteria. NRC-approved or industry-accepted computer codes and calculational techniques were used to perform the calculations that demonstrate meeting the acceptance criteria. Similarly, design margins specified by application of the ASME design rules are maintained, as are other margin-ensuring criteria used to judge the acceptability of the plant. Environmental margins are maintained by not increasing any of the present limits for releases, such as ultimate heat sink maximum temperature or plant vent radiological limits.

11.4.2.3 Fuel Thermal Limits

No change is required in the basic fuel design to achieve the EPU RTP level or to meet the plant licensing limits. No increase in allowable peak bundle power is requested for the EPU. Analyses for each fuel reload will continue to meet the criteria accepted by the NRC as specified in NEDO-24011 (GESTAR II) or otherwise approved in the Technical Specifications. No new fuel design is required for the EPU. Plus, future fuel designs will meet acceptance criteria approved by the NRC.

11.4.2.4 Makeup Water Sources

The BWR design concept includes a variety of ways to pump water into the reactor vessel to deal with all types of events. There are numerous safety-related and nonsafety-related cooling water sources. The safety-related cooling water sources alone can maintain core integrity by providing adequate cooling water. There are high and low pressure, high and low volume, safety and non-safety grade means of delivering water to the vessel. These means include the feedwater and condensate system pumps, the low pressure emergency core cooling systems (ECCS) (Low Pressure Coolant Injection (LPCI) and Core Spray (CS)) pumps, the high pressure ECCS (High Pressure Coolant Injection (HPCI)) pump, the Reactor Core Isolation Cooling (RCIC) pump, the Standby Liquid Control (SLC) pumps, and the Control Rod Drive (CRD) pumps.

The EPU does not result in an increase or decrease in the available water sources, nor does it change the selection of those assumed to function in the safety analyses. NRC-approved methods were used for analyzing the performance of the ECCS during Loss-of-Coolant Accidents (LOCAs).

The EPU results in a 14.3% increase in decay heat, and thus, the time to reach cold shutdown increases. This is not a safety concern, and the existing cooling capacity can bring the plant to cold shutdown within an acceptable time span.

11.4.2.5 Design Basis Accidents

Design Basis Accidents (DBAs) are very low probability hypothetical events whose characteristics and consequences are used in the design of the plant, so that the plant can mitigate their consequences to within acceptable regulatory limits. For BWR licensing evaluations, capability is demonstrated for coping with the range of hypothetical pipe break sizes in the largest recirculation, steam, and feedwater lines, a postulated break in one of the ECCS lines, and the most limiting small lines, while accommodating a single active equipment failure. This

break range bounds the full spectrum of large and small, high and low energy line breaks. Several of the most significant licensing assessments are made using these LOCA ground rules. These assessments are:

- Challenges to Fuel (ECCS Performance Analyses) in accordance with the rules and criteria of 10 CFR 50.46 and Appendix K wherein the predominant criterion is the fuel peak clad temperature (PCT).
- Challenges to the Containment wherein the primary criteria of merit are the maximum containment pressure calculated during the course of the LOCA and maximum suppression pool temperature for long-term cooling in accordance with 10 CFR 50 Appendix A Criterion 38.
- DBA Radiological Consequences calculated and compared to the criteria of 10 CFR 50.67.

11.4.2.6 Challenges to Fuel

The ECCS are described in Section 6.3 of the plant Updated Final Safety Analysis Report (UFSAR). The ECCS Performance Evaluation was conducted through application of the 10 CFR 50 Appendix K evaluation models, and demonstrates the continued conformance to the acceptance criteria of 10 CFR 50.46. As mentioned above, a complete spectrum of pipe breaks was investigated from the largest recirculation line down to the most limiting small line break. The licensing safety margin is not affected by the EPU. The increased PCT consequences for the EPU are insignificant compared to the large margin to the regulatory criteria. Therefore, the ECCS acceptance criteria continue to be satisfied

11.4.2.7 Challenges to the Containment

The effect of the EPU on the peak values for containment pressure and temperature confirms the suitability of the plant for operation at the EPU RTP level. Also, the effects of the EPU on the conditions that affect the containment dynamic loads are determined, and the plant is judged satisfactory for EPU power operation. Where plant conditions with the EPU are within the range of conditions used to define the current dynamic loads, current safety criteria are met and no further structural analyses is required.

11.4.2.8 Design Basis Accident Radiological Consequences

The UFSAR provides the radiological consequences for each DBA. The magnitude of the potential consequences is dependent upon the quantity of fission products released to the

environment, the atmospheric dispersion factors and the dose exposure pathways. The dose exposure pathways for the EPU and Alternative Source Term (AST) implementation are updated to include (1) consideration of a positive pressure period in the secondary containment that results in leakage to the environment until a negative pressure is re-established, and (2) secondary containment bypass leakage via main steam isolation valve (MSIV) leakage, and MSIV, RCIC, and HPCI drains into the condenser. The atmospheric dispersion factors have been revised for AST implementation. The quantity of fission products is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point.

For the EPU, the Control Rod Drop Accident (CRDA), LOCA, Fuel Handling Accident (FHA) and the Main Steam Line Break Accident (MSLBA) were reanalyzed, and the instrument line break accident (ILBA) was reviewed.

For the MSLBA, the quantity of activity in the primary coolant and in the offgas used in the evaluation of this postulated event is based on Technical Specification limits, which remain unchanged for the EPU/AST. The EPU/AST updated MSLBA analysis is affected by the implementation of the AST, which includes the Technical Specification limit activity and the revised atmospheric dispersion factors.

For the ILBA, the only transport mechanism influenced by the EPU is the quantity of coolant mass discharged to the environment. The ILBA is not a limiting event. For the ILBA, increased mass loss would only occur if the operating pressure were increased. However, the requested EPU does not need or include an increase in operating pressure, and thus, the consequences of an ILBA do not change.

For the remaining DBAs (i.e., CRDA, LOCA, and FHA), the only parameter of importance is the activity released from the fuel. Because the mechanism of fuel failure is not influenced by the EPU, the only parameter of importance is the actual inventory of fission products in the fuel rod. Because the only parameter affecting fuel is an increase in thermal power, the increase in the quantity of fission products can be assumed to be proportional to the increase in power.

The DBA, which has historically been limiting from a radiological viewpoint, is the LOCA, for which Regulatory Guide 1.183 has been applied. For this accident, the BWR AST release fractions specified in Regulatory Guide 1.183 are assumed. These release fractions are not influenced by the EPU. The radiological consequences from the updated EPU/AST LOCA

DBA, as shown in Section 9, remain below regulatory guidelines. The EPU LOCA evaluation results include the 2% power uncertainty factor from Regulatory Guide 1.49.

The results of all radiological analyses remain below the 10 CFR 50.67 guideline values. Therefore, radiological safety margins are maintained.

11.4.2.9 Transient Analyses

The effects of plant transients are evaluated by investigating a number of disturbances of process variables and malfunctions or failures of equipment according to a scheme of postulating initiating events. These events are primarily evaluated against the Safety Limit Minimum Critical Power Ratio (SLMCPR). The Operating Limit MCPR is increased appropriately to assure that the SLMCPR is not infringed upon, if any transient is initiated from the EPU RTP level. The limiting transients are analyzed for each specific fuel cycle. Licensing acceptance criteria are not exceeded.

11.4.2.10 Combined Effects

The EPU analyses use fuel designed to current NRC-approved criteria and are operated within NRC-approved limits to produce more power in the reactor, and thus, increase steam flow to the turbine. NRC-approved design criteria are used to assure equipment mechanical performance at the EPU conditions. Scram frequency is maintained by small adjustments to reactor instrumentation. These adjustments are attributed to the small changes in the reactor operating conditions. DBAs are hypothesized to evaluate challenges to the fuel, containment and off-site dose limits. These challenges have been evaluated separately in accordance with extremely conservative regulatory procedures such that the separate effects are more severe than any combined effects. The off-site dose evaluation specified by Regulatory Guide 1.183 and SRP-15.6.5 provides a more severe DBA radiological consequences scenario than the combined effects of the hypothetical LOCA, which produces the greatest challenge to the fuel and/or containment. That is, the DBA, which produces the highest PCT and/or containment pressure, does not damage large amounts of fuel, and thus, the source terms and doses are much smaller than those postulated in evaluations conforming to Regulatory Guide 1.183.

11.4.2.11 Non-LOCA Radiological Release Accidents

All of the limiting non-LOCA events discussed in Regulatory Guide 1.70 Chapter 15 have been updated for the effect of the EPU/AST. The dose consequences for all of the non-LOCA radiological release accident events are shown in Section 9 to remain below regulatory limits.

11.4.2.12 Equipment Qualification

Plant equipment and instrumentation has been evaluated against the criteria appropriate for the EPU. Significant groups/types of equipment have been justified for the EPU by generic evaluations. Some of the qualification testing/justification at the current RTP level was done at more severe conditions than the minimum required. In some cases, the qualification envelope did not change significantly due to the EPU. A process has been developed to ensure qualification of the equipment whose current qualification does not already bound the EPU conditions.

11.4.2.13 Balance-of-Plant

Balance-of-plant (BOP) systems/equipment used to perform safety-related and normal operation functions have been reviewed for the EPU in a manner comparable to that for safety-related NSSS systems/equipment. This includes, but was not necessarily limited to, all or portions of the Main Steam, Feedwater, Turbine, Condenser, Condensate, Service Water, Emergency Diesel Generator, BOP piping, and support systems. Significant groups/types of BOP equipment/systems are justified for the EPU by generic evaluations. Plant-specific evaluations justify EPU operation for BOP systems/equipment that are not generically justified.

11.4.2.14 Environmental Consequences

The non-radiological environmental effects resulting from the BSEP EPU will be minimal. The only effect of consequence will be a relatively small increase in the amount of waste heat discharged to the Atlantic Ocean. CP&L has submitted an application, to the State of North Carolina, for a revision to its NPDES permit. This revision, in part, adjusts the established ocean discharge mixing zone to account for the slight increase in circulating water temperatures, thermal limits established in the NPDES permit are not revised. The overall effect of this discharge temperature increase on ocean temperature is negligible.

There will be no significant radiological environmental effects resulting from the BSEP EPU. There will be no change in the quantity of radioactivity released to the environment through liquid effluents, and only a small increase in airborne emissions of radioactivity. All offsite radiation doses will be small and within 10 CFR 20 and 10 CFR 50, Appendix I limits.

As a result, it is concluded that the BSEP EPU does not constitute an unreviewed environmental question and that the BSEP EPU is eligible for categorical exclusion as provided by 10 CFR 51.22(c)(9).

11.4.2.15 Technical Specification Changes

The Technical Specifications ensure that plant process variables and system performance parameters are maintained within the values assumed in the safety analyses. That is, the Technical Specification parameters (process variables, Allowable Values, operating limits, etc.) are selected such that the actual equipment is maintained equal to or more conservative than the assumptions used in the safety analyses. The Technical Specification changes justified by the safety analyses summarized in this report are listed in Table 11-1.

Proper account is taken of inaccuracies introduced by instrument drift, instrument accuracy, and calibration accuracy. For example, to assure conservatism in a high reactor pressure safety analysis event, the high reactor pressure trips are set lower in the Technical Specifications than those used in the safety analysis. This assures that the actual plant responses will be less severe than those represented by the safety analysis. Similarly, the Technical Specifications address equipment operability (availability) and put limits on equipment out-of-service (not available for use) times such that the actual plant can be expected to have at least the complement of equipment available to mitigate abnormal plant events assumed in the safety analyses. Because the safety analyses for the EPU show that the results are acceptable within regulatory limits, public health and safety is confirmed. Technical Specification changes consistent with the EPU RTP level are made in accordance with methodology already approved for the plant and continue to provide a comparable level of protection as Technical Specifications previously issued by the NRC.

11.4.3 Assessment Against 10 CFR 50.92 Criteria

10 CFR 50.91(a) states "At the time a licensee requests an amendment, it must provide to the Commission ... its analysis about the issue of no significant hazards consideration using the standards in § 50.92." The following provides this analysis for the BSEP 1 and 2 120% of OLTP EPU. The conclusions are based on the evaluations provided in this report, and are summarized as appropriate to the following safety considerations in accordance with 10 CFR 50.92.

1) Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

As summarized below, the increase in power level discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of Design Basis Accidents is not affected by the increased power level, because the plant still complies with the regulatory and design basis criteria established for plant equipment (e.g., ASME code, IEEE standards, NEMA standards, Regulatory Guide criteria). An evaluation of the BWR probabilistic risk assessments concludes that the calculated core damage frequencies do not significantly change due to the EPU. Instrument setpoints (equipment settings that initiate automatic plant trips) and equipment operating margins are established such that there is no expected increase in transient event frequency due to the EPU. No new challenges to safety-related equipment result from the EPU.

Radiological release events (accidents) have been evaluated, and shown to meet the regulatory limits of 10 CFR 50.67. In all cases, the consequences of hypothetical accidents, compared to those previously evaluated, are not significantly increased. The spectrum of hypothetical accidents and transients has been investigated, and are shown to meet existing regulatory limits.

Challenges to the major fission product barriers: fuel cladding, reactor coolant pressure boundary, and containment have all been evaluated.

Challenges to the fuel cladding from abnormal transients and accidents have been analyzed and appropriate limits established (e.g., Maximum Average Planar Linear Heat Generation Rate [MAPLHGR] and SLMCPR) to ensure that fuel cladding integrity will be maintained under EPU conditions.

Challenges to the Reactor Coolant Pressure Boundary were evaluated under EPU conditions (pressure, temperature, flow and radiation) and found to meet their acceptance criteria for allowable stresses and overpressure margin.

Challenges to the containment under postulated EPU accident conditions have been evaluated, and the containment and its associated cooling systems continue to demonstrate margin to their design basis pressure and temperature limits.

Based on the above, it is concluded that implementation of the EPU will not significantly increase either the probability or consequences of any previously evaluated accident.

2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

As summarized below, EPU implementation will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Equipment that could be affected by the EPU has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario or equipment failure mode was identified. The full spectrum of accident considerations, defined in the UFSAR, has been evaluated, and no new or different kind of accident has been identified. EPU implementation uses already developed technology and NRC-approved safety analysis methodology, and applies them within the capabilities of already existing plant equipment in accordance with presently existing regulatory and industry criteria. GE has designed BWRs of higher power levels than the EPU RTP level of any operating BWR in the fleet, and no new power dependent accidents have been identified.

The Technical Specification changes needed to implement the EPU require some in-plant adjustments, but no change to the plant's physical configuration. All changes have been evaluated, are acceptable, and do not create any new accident scenarios.

3) Will the change involve a significant reduction in a margin of safety?

As summarized below, this change will not involve a significant reduction in a margin of safety.

Challenges to the fuel, reactor coolant pressure boundary, and containment were reanalyzed for EPU conditions. The fuel integrity is maintained by meeting existing design and regulatory limits. The challenges to all affected structures, systems and components have been evaluated and will remain within their acceptance criteria for all design basis events. Therefore, although some design and operational margins are affected by the EPU, the margins of safety currently designed into the plant are not significantly affected.

Because the plant response to transients and hypothetical accidents does not result in exceeding any NRC regulatory limits, EPU implementation does not involve a significant reduction in a margin of safety.

Conclusions:

An EPU to 120% of OLTP has been investigated. The method for achieving higher power is to slightly increase some plant operating parameters. The challenges to plant systems, structures, and components have been evaluated and demonstrate how this EPU can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without exceeding any presently existing regulatory limits applicable to the plant which might cause a reduction in a margin of safety.

Having arrived at negative declarations with regards to the criteria of 10 CFR 50.92, this assessment concludes that the EPU of the amount described herein does not involve a Significant Hazards Consideration.

Table 11-1

Technical Specifications and Bases Affected by EPU

TS Item	Description of Change
Section 1.1, Definitions	Revised the definition of RATED THERMAL POWER to be the EPU maximum licensed power level of 2923 MWt.
Safety Limit (SL) 2.1.1.1	Revised the SL for fuel cladding integrity at low core flow and reactor pressure from 25% RTP to 23% RTP.
Limiting Condition for Operation (LCO) 3.1.3: - Condition D; LCO 3.1.6: - Applicability; LCO 3.3.2.1: - Surveillance Requirement (SR) 3.3.2.1.2 - SR 3.3.2.1.3 - SR 3.3.2.1.5 - Table 3.3.2.1-1, Note (f)	Revised the applicable THERMAL POWER from 10% RTP to 8.75% RTP.
LCO 3.2.1: - Applicability - Req'd Action B.1 - SR 3.2.1.1 LCO 3.2.2: - Applicability - Req'd Action B.1 - SR 3.2.2.1	Revised the percentage of RTP value related to thermal limits monitoring from 25% RTP to 23% RTP.
LCO 3.3.1.1: - SR 3.3.1.1.3	Revised the percentage of RTP value contained in the SR and the associated NOTE from 25% RTP to 23% RTP. This value establishes the minimum power level at which the average power range monitors (APRM) are adjusted to conform to the calculated power.
LCO 3.3.1.1: - Required Action E.1 - SR 3.3.1.1.16 - Table 3.3.1.1-1, Functions 8 and 9	Revised the percentage of RTP value from 30% RTP to 26% RTP. This value corresponds to the power level at which the Turbine Stop Valve (TSV) closure and Turbine Control Valve (TCV) fast closure trips of the Reactor Protection System (RPS) are bypassed.

TS Item	Description of Change
LCO 3.3.1.1: - Table 3.3.1.1-1, Function 2b Footnote (b)	Revised the allowable value for the APRM Simulated Thermal Power - High from 0.66W + 62.0 %RTP to 0.55W + 62.6 %RTP For single loop operation, revised the allowable value for the APRM Simulated Thermal Power - High from 0.66(W - ΔW) + 62.0 %RTP to 0.55(W - ΔW) + 62.6 %RTP
LCO 3.3.2.2 - Applicability - Required Action C.1	Revised the percentage of RTP value at which the Feedwater and Main Turbine High Water Level Trip Instrumentation is required OPERABLE from 25% RTP to 23% RTP.
LCO 3.7.6: - Applicability - Required Action B.1	Revised the percentage of RTP value at which the Main Turbine Bypass Valve system is required OPERABLE from 25% RTP to 23% RTP.

12 REFERENCES

1. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," Licensing Topical Report NEDO-32424, Class I (Non-proprietary), April 1995.

2. BWROG-94078, "BWR Owner's Group Guidelines for Stability Interim Corrective Action," June 1994.

3. GE Nuclear Energy, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," Licensing Topical Report NEDO-32465-A, August 1996.

4. NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE CODE)," April 1976.

5. "Alternative Radiological Source Terms for Evaluating the Radiological Consequences of Design Basis Accidents at Boiling and Pressurized Water Reactors," Regulatory Guide 1.183, July 2000.