GE Nuclear Energy

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September 1995

Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2

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San Jose, CA 95125

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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EXECUTIVE SUMMARY

Licensing Approach

This report summarizes the evaluations performed that justify uprating the licensed thermal power at Brunswick Steam Electric Plant by 5% to 2558 MWt.

This report uses the NRC-approved generic format and content for BWR power uprate licensing reports documented in NEDC-31897P-A, *Generic Guidelines For General Electric Boiling Water Reactor Power Uprate,* commonly called "LTR1". Per LTR1, every issue that should be addressed in a plant-specific power uprate licensing report is addressed in this report. For issues that are evaluated generically, this report usually only references the generic evaluations in either LTR1 or NEDC-31984P, *Generic Evaluations of General Electric Boiling Water Reactor Power Uprate,* which is commonly called "LTR2".

It is not the intent of this report to address all the details of the evaluations reported herein. For example, only previously NRC-approved methods were used for all the analyses of accidents and transients, as documented in LTR1. Therefore, safety analysis methods have been previously addressed, and need not be addressed in this report. Also, event and analysis descriptions that are already provided in other licensing reports or the Updated Final Safety Analysis Report are not repeated within this report. This report summarizes the results of all the significant operational and safety evaluations needed to justify a licensing amendment to allow for uprated power operation.

Uprating the power level of nuclear power plants can be done safely within certain plant specific limits. Many light water reactors, including several BWR plants, have already been uprated world wide.

An increase in electrical output of a BWR plant is accomplished primarily by generation and supply of higher steam flow for the turbine generator. Brunswick Units 1 and 2, like most BWR plants, as originally licensed have an as-designed equipment and system capability to accommodate steam flow rates at least 5% above the original rating.

The predominant plant licensing challenges have been reviewed, and this uprate 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 plants, which might cause a significant reduction in a margin of safety.

Uprate of the amount described herein involves no significant hazards consideration for either Brunswick Unit 1 or 2.

Licensing Criteria

The evaluations and reviews for uprating the Brunswick units to 2558 MWt were conducted in accordance with the criteria in Appendix B of NEDC-31897P-A. The results of these evaluations and reviews are presented in the succeeding sections of this report.

- All safety aspects of the plants that are affected by the increase in thermal power, including nuclear steam supply and balance-of-plant systems, were evaluated.
- There is no change in the established licensing basis of the plants.
- Evaluations were performed using NRC-approved analysis methods.
- No changes to comply with more recent codes and standards are being requested.
- The Nuclear Steam Supply System was reviewed to confirm that it continues to comply with the Updated Final Safety Analysis Report.
- No modifications (other than instrument setpoints) are expected; however, any modifications will be implemented per 10CFR50.59.
- All systems and components impacted by power uprate were reviewed to ensure there are no significant challenges to safety systems.
- Compliance with existing plant environmental regulations was reviewed.
- A review as defined in 10CFR50.92(c) was performed.
- Changes to systems since the original operating license have been evaluated. Changes not yet implemented have also been reviewed for the effects of power uprate.

The 105% power uprate is accomplished with an associated increase in vessel steam flow to approximately 106% of steam flow at the current licensed power level.

1.0 OVERVIEW

1.1 Introduction

Uprating the power level of nuclear power plants can be done safely within certain plant specific limits. Most GE BWR plants have the capability and margins for an uprating of 5 to 20% without major hardware modifications. Many light water reactors, including several BWR plants, have already been uprated world wide. Over a thousand MWe have already been added by uprate in the United States. This evaluation justifies a power uprate to 2558 MWt, which corresponds to 105% of the previously rated thermal power for Brunswick Units 1 and 2.

See Table 1-1 for a glossary of terms used in this report.

1.2 Purpose and Approach

An increase in electrical output of a BWR plant is accomplished primarily by generation and supply of higher steam flow to the turbine generator. Most BWRs, including Brunswick Units 1 and 2 (as originally 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 (computer codes) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core designs have resulted in a significant increase in the difference between the calculated safety analyses results and the licensing limits. These available safety difference calculational results, combined with the as-designed excess equipment, system, and component capabilities, provide BWR plants with the potential for an increase in their thermal power rating of 5% and more without major nuclear steam supply system (NSSS) hardware modifications and with no significant increase in the hazards presented by the plant as approved by the NRC at the original license stage.

The uprate parameters specific to the Brunswick units are provided in Table 1-2.

The method for achieving higher power is to extend the power/flow map by increasing core flow along existing flow control lines. However, there will not be an increase in the maximum recirculation flow limit over the pre-uprate value. Uprated operation will also involve slightly higher reactor vessel dome pressure to provide adequate inlet pressure conditions at the turbine, accounting for the larger pressure drop through the steam lines at higher flow, and to provide sufficient pressure control and turbine flow capability.

1.2.1 Uprate Analysis Basis

The Brunswick plants were originally licensed at 2436 MWt. The original safety analysis basis assumed that the reactor had been operating continuously at a power level at least 1.02 times the licensed power level; many of the original analyses had already been performed at 105% steam flow. The uprate power level included in this evaluation is 105% of the original licensed power level. The power uprate safety analyses are based on a power level of at least 1.02 x 1.05 x (original licensed power level), which equals at least 1.071 x (original licensed power level), or at least 1.02 x (uprated power level).

Uprate	Multiple of Rated Power	Power	Analysis***
0	1.0	X*	≥1.02X
5%	1.05	1.05X = Y	≥1.071X
5%	1.05	Y**	≥1.02Y

* Original power level (2436 MWt)

** Uprated power level (2558 MWt)

*** Some analyses are performed at 100% rated power, because the Regulatory Guide 1.49 2% power factor is already accounted for in the analysis methods.

1.2.2 Margin

The above described uprate analysis basis assures that the power-dependent margin prescribed by the Code of Federal Regulations (CFR) will be maintained by meeting the appropriate regulatory criteria. NRC-accepted computer codes and calculational techniques are used to make the calculations that demonstrate meeting the stipulated criteria. Similarly, margins specified by application of the code design rules will be maintained, as will other margin-assuring acceptance criteria used to judge the acceptability of the plant.

1.2.3 Approach

The planned approach to achieving the higher power level consists of: (1) an increase in the core thermal power to create increased steam flow, (2) a corresponding increase in the feedwater system flow, (3) no increase in maximum core flow, and (4) reactor operation primarily along extensions of current rod/flow control lines. This approach is based on and is consistent with the NRC-approved BWR generic power uprate guidelines presented in NEDC-

31897P-A (Reference 1).

1.3 Uprated Plant Operating Conditions

The following evaluations justify increasing the rated thermal power to 105% of original values. The following descriptions provide information on the original and uprated plant operating conditions.

1.3.1 Reactor Heat Balance

The thermal-hydraulic performance of a BWR reactor core is characterized by the total operating power, operating pressure, total core flow, and 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 power uprate conditions.

The reactor heat balance relates the thermal-hydraulic parameters to the plant steam and feedwater flow conditions for the selected core thermal power level and operating pressure.

Operational parameters from actual plant operation are considered (e.g., steam line pressure drop) to match expected uprate conditions. The thermal-hydraulic parameters define the conditions for evaluation of operation of the plant at uprate power. The thermal-hydraulic parameters obtained for the uprated power also define the steady-state operating conditions for equipment evaluations. Heat balances at appropriately selected conditions define the initial and boundary conditions for plant safety analyses.

The uprated heat balance is based on a thermal power increase of 5%. The reactor heat balance is coordinated with the turbine heat balance. Figure 1-1 shows the uprated reactor heat balance at 100% of uprated power and 100% rated core flow that was utilized as the analysis basis for both units.

1.3.2 Reactor Performance Improvement Features

1.4 Summary and Conclusions

This evaluation has investigated power uprate at 105% of original rated thermal power. The strategy for achieving higher power is to expand the power/flow map by increasing core flow along existing flow control lines. The predominate plant licensing challenges have been reviewed to demonstrate how this uprate 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 significant reduction in a margin of safety. A power uprate of the amount described herein involves no significant hazard consideration.

1.5 References

1. GE Nuclear Energy, *Generic Guidelines For General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDO-31897, Class I (Non-proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.

2. GE Nuclear Energy, *Generic Evaluations of General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-proprietary), March 1992; and Supplements 1 & 2.

Table 1-1
GLOSSARY OF TERMS

Term	Definition
ADS	Automatic Depressurization System
AOO	Anticipated Operating Occurrences (Moderate Frequency Transient Events)
APLHGR	Average Planar Linear Heat Generation Rate
APRM	Average Power Range Monitor
ARO	All Rods Out
ARTS	APRM/RBM/Technical Specifications
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BHP	Brake Horse Power
BOP	Balance of Plant
BWR	Boiling Water Reactor
CC	Containment Spray
CFD	Condensate Filter Demineralizers
CRD	Control Rod Drive
DAR	Design Assessment Report
DBA	Design Basis Accident
DG	Diesel Generator
DGSWS	Diesel Generator Service Water System
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generators
EECW	Emergency Equipment Cooling Water
EESW	Emergency Equipment Service Water
EHC	Electrohydraulic Control
ELLL	Extended Load Line Limit
EOOS	Equipment Out-of-Service
EQ	Environmental Qualification
FES	Final Environmental Statement
FFWTR	Final Feedwater Temperature Reduction
FSAR	Final Safety Analysis Report
FWCF	Feedwater Controller Failure
FWHOS	Feedwater Heater(s) Out-of-Service
GE	General Electric Company
SW	Service Water
HCU	Hydraulic Control Unit
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HPCI/S	High Pressure Coolant Injection/Core Spray
ICF	Increased Core Flow
IEB	Inspection and Enforcement Bulletin

Table 1-1

GLOSSARY OF TERMS

(Continued)

Term	Definition
IGSCC	Intergranular Stress Corrosion Cracking
LCS	Leakage Control System
LFA	Lead Fuel Assemblies
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
LTP	Long-Term Plan
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELB	Moderate Energy Line Break
MELLL	Maximum Extended Load Line Limit
MEOD	Maximum Extended Operating Domain
MG	Motor Generator
MSIV	Main Steam Isolation Valve
MSR	Moisture Separator Reheater
MWt/MWth	Megawatt-thermal
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUREG	Nuclear Regulations
OLMCPR	Operating Limit Minimum Critical Power Ratio
PCS	Pressure Control System
PCT	Peak Cladding Temperature
PUAR	Plant Unique Analysis Report
PULD	Plant Unique Load Definition
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RFP	Reactor Feed Pump
RHR	Residual Heat Removal
RHRSWS	Residual Heat Removal Service Water System
RIPD	Reactor Internals Pressure Difference
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RT_{NDT}	Reference Temperature of Nil-Ductility Transition

Table 1-1

GLOSSARY OF TERMS

(Continued)

Term	Definition
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
SAR	Safety Analysis Report
SBO	Station Blackout
SGTS	Standby Gas Treatment System
SIL	Services Information Letter
SJAE	Steam Jet Air Ejectors
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-Loop Operation
SPCM	Suppression Pool Cooling Mode
SRM	Source Range Monitor
SRV	Safety/Relief Valve
SRVDL	Safety/Relief Valve Discharge Line
STPM	Simulated Thermal Power Monitor
TB	Turbine Bypass
TBCCW	Turbine Building Closed Cooling Water System
TCV	Turbine Control Valve
TIP	Traversing In-core Probe
TSV	Turbine Stop Valve
TG	Turbine Generator
TGT	Turbine Generator Trip
TPM	Thermal Power Monitor
USAR/UFSAR	Updated Safety Analysis Report/Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USE	Upper Shelf Energy

Table 1-2

ORIGINAL AND UPRATED PLANT OPERATING CONDITIONS

Parameter	Current* Rated Power Value	Uprated Power Value
Thermal Power (MWt)	2436	2558
Vessel Steam Flow (Mlb/hr)**	10.470	11.077
Full Power Core Flow Range		
Mlb/hr	57.8 to 80.8	62.4 to 80.3
% Rated	75.0 to 105.0	81 to 104.3***
Dome Pressure (psia)	1020	1045
Dome Temperature (°F)	547	550
Turbine Inlet Pressure (psia)	965	1000
Full Power Feedwater		
Flow (Mlb/hr)	10.440	11.054
Temperature (°F)	420	425
Core Inlet Enthalpy (Btu/lb)	526.9	529.7

* Based on Unit 1 Cycle 8 fuel reload analysis. These values are bounding for Unit 2, except as noted.
** At 100% core flow condition.
***For Unit 2, the full power core flow range is 62.4 to 80.5 Mlb/hr, 81% to 104.5% of rated.

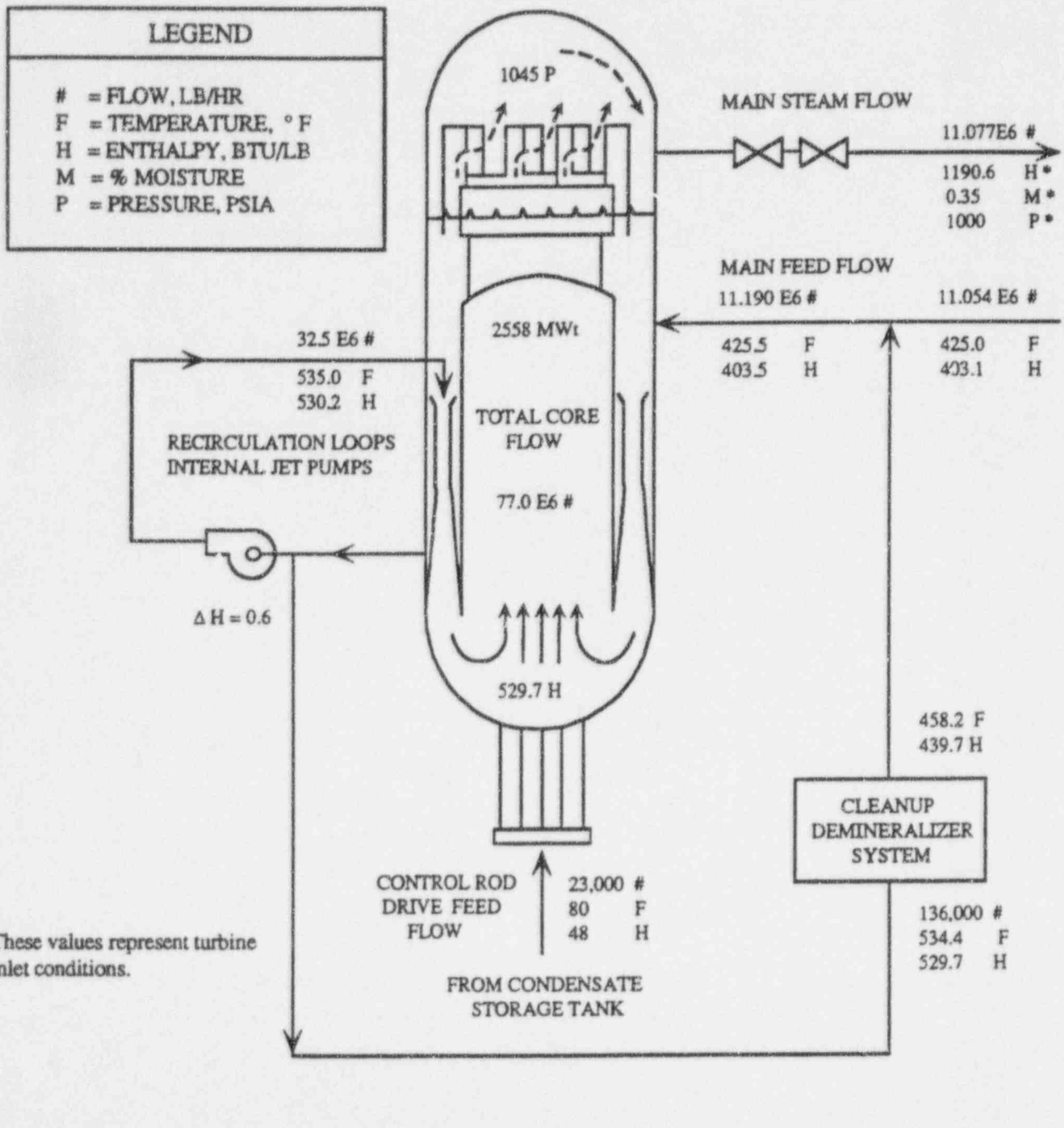

Figure 1-1. Uprated Heat Balance (Nominal) 100% Power and 100% Core Flow

2.0 REACTOR CORE AND FUEL PERFORMANCE

This section focuses primarily on the information requested in Regulatory Guide 1.70, Chapter 4, that applies to power uprate.

2.1 Fuel Design and Operation

At original or uprated conditions, all fuel and core design limits will continue to be met by control rod pattern adjustments. New fuel designs are not needed for power uprate to assure adequate safety. The increased energy requirements associated with power uprate will be accommodated by an increase in the reload fuel enrichment. Thus, the number of fuel assemblies requiring disposal is not expected to increase. Furthermore, improvements in fuel bundle efficiency will offset the increased fuel enrichments associated with power uprate. The reload core for power uprate operation has not been designed yet, so the degree of offset has not been determined. Power uprate may result in an increase in fuel burnup relative to the current level of burnup, but NRC-approved limits on the fuel designs to be utilized will not be exceeded. The impact of higher power operation on radiation sources and design basis accident doses are discussed in Sections 8.3 and 9.2, respectively.

2.2 Thermal Limits Assessment

For the subjects to be addressed in Sections 2.2.1 and 2.2.2, the plant is bounded by the generic evaluations provided in NEDC-31984P (Reference 1).

2.2.1 Minimum Critical Power Ratio (MCPR) Operating Limit

The operating limit MCPR is determined on a cycle-specific basis from the results of the reload transient analysis, as described in Section 5.3 of Reference 1. This approach does not change for power uprate.

2.2.2 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Maximum Linear Heat Generation Rate (LHGR) Operating Limits

Operation within the MAPLHGR and LHGR limits will be maintained as described in Section 5.1 of Reference 1. This approach does not change for power uprate.

2.3 Reactivity Characteristics

2.3.1 Power/Flow Operating Map

The uprated power/flow operating map (Figure 2-1) includes the operating domain changes for uprated power and the plant performance improvement features addressed in Section 1.3.2 The changes to the power/flow operating map are consistent with the NRC-approved generic descriptions given in Sections 5.2 and C.2.3 of NEDC-31897P-A (Reference 2). The maximum thermal operating power and maximum core flow shown on Figure 2-1 correspond to the uprated power and the previously analyzed core flow range when rescaled so that uprated power is equal to 100% rated.

2.4 Stability

To maintain the same level of stability during uprated operation, the plant will be operated as described in Section 3.2 of Reference 1. CP&L's commitments to short and long-term resolution of the stability issue are unaffected by uprate. The design of Enhanced Stability Option I-A, which will be implemented by the Brunswick units to address the stability issue, will incorporate the power/flow map and applicable instrumentation setpoints associated with power uprate operation.

2.5 Reactivity Control

2.5.1 Control Rod Drive Hydraulic System

The Control Rod Drive (CRD) System controls gross 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.

The increase in dome pressure due to power uprate produces a corresponding increase in the bottom head pressure. Initially, rod insertion is slowed slightly due to increased pressure. A maximum scram time increase of 0.006 seconds is anticipated for 5% rod insertion. At approximately 10-15% insertion, the scram insertion times are approximately the same as for the current power level. As the scram continues, reactor pressure becomes the primary source of pressure to complete the scram. Hence, the higher reactor pressure will improve scram performance after approximately 20% rod insertion. Therefore, an increase in the reactor pressure has little effect on overall scram time (less than 6 milliseconds), and CRD performance during power uprate will meet current Technical Specification scram time requirements.

Based on the above, the CRD System will continue to carry out its functions at uprated power.

2.5.1.1 Control Rod Drive Mechanism

Section 2.5.1 states that the CRD mechanism structural and functional integrity is acceptable for at least 1250 psig (which is above the uprate operating pressure and the high pressure scram setpoint, including hydrostatic head).

The components of the CRD mechanism designated as a primary pressure boundary have been designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The applicable ASME Code effective date for the initially supplied Brunswick Unit 1 CRDs is the 1968 Edition up to and including the Winter 1969 addenda. The applicable ASME Code effective date for the initially supplied Brunswick Unit 2 CRDs is the 1968 Edition up to and including the Winter 1968 addenda.

The limiting component of the CRD mechanism is the indicator tube, which has a calculated stress of 20,790 psi; the allowable stress is 26,060 psi. The maximum stress on this component results from [illegible] maximum CRD internal hydraulic pressure of 1750 psig.

The cyclic operation of the CRD was conservatively evaluated in accordance with ASME Code N-415.1. All requirements of N-415.1 are satisfied even when considering the increased power uprate vessel bottom head pressure, thereby satisfying the peak stress intensity limits governed by fatigue. The limiting component was found to be the CRD main flange. The fatigue usage factor is 0.15, which is less than the allowable limit of 1.0.

The CRD mechanism has been subjected to intensive testing at 1250 psig, which is higher than the maximum power uprate high pressure scram setpoint. Based on the demonstrated performance of the mechanism at these high pressures, it is concluded that deformation resulting from the power uprate pressure increase is of no significant consequence.

2.6 References

1. GE Nuclear Energy, *Generic Evaluations of General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-Proprietary), March 1992; and Supplements.

2. GE Nuclear Energy, *Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDO-31897, Class I (Non-proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.

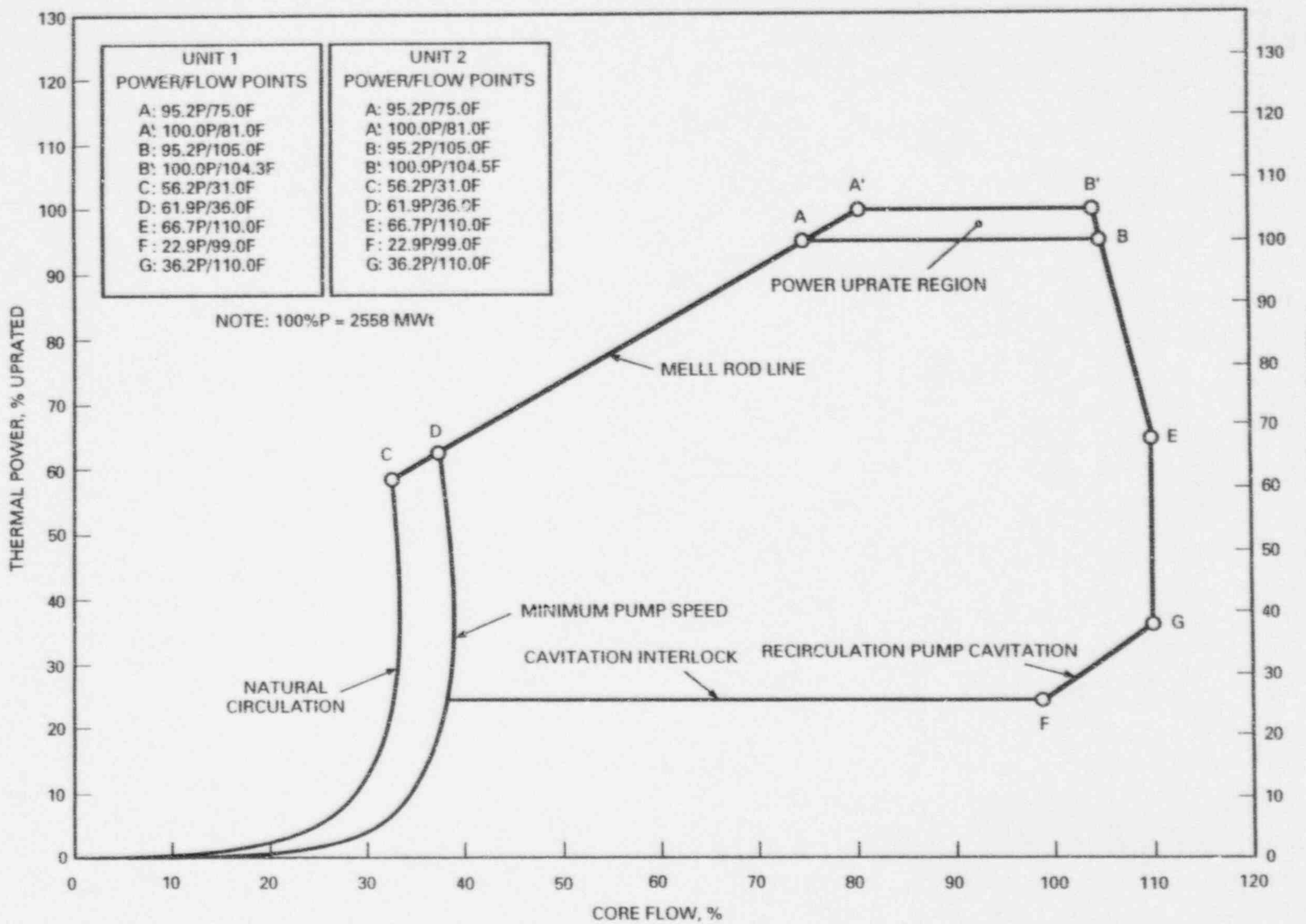

Figure 2-1. Power/Flow Operating Map for Power Uprate

3.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

This section focuses primarily on the information that is requested in Regulatory Guide 1.70, Chapters 3 and 5, that applies to power uprate.

3.1 Nuclear System Pressure Relief

The purpose of the nuclear system pressure relief function is to prevent overpressurization of the nuclear system during abnormal operational transients. The plant safety/relief valves (SRVs), along with scram, provide this protection.

Additional discussions of the SRVs are provided in Subsection 4.1.2.2 of this report and Section 4.6 of NEDC-31984P (Reference 1).

3.2 Reactor Overpressure Protection

The design pressure of the Brunswick Units 1 and 2 reactor vessel and reactor pressure coolant boundary is 1250 psig. The ASME Code allowable peak pressure is 1375 psig (110% of design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a main steamline isolation valve (MSIV) closure with flux scram, which is described in Section S.1.2 of NEDE-24011-P-A-10-US (Reference 2). Two SRVs out-of-service (OOS) were assumed in the overpressure protection analysis for consistency with previous analyses. The initial reactor dome pressure used in the overpressure protection analysis was 1060 psia. The analysis did not take credit for relief flow. At uprated conditions, the peak RPV pressure increases by 55 psi to 1344 psig, but remains below the 1375 psig ASME limit.

Therefore, there is no decrease in margin of safety. The results of the uprate overpressure protection analysis are given in Figure 3-1.

3.3 Reactor Vessel and Internals

A comprehensive review of the effects of increased power and pressure conditions on the reactor vessel and internals was completed.

3.3.1 Reactor Vessel Fracture Toughness

Reactor pressure vessel (RPV) embrittlement is caused by neutron exposure of the wall adjacent to the core (the "beltline" region).

Because operation with power uprate is projected to remain within 32 EFPY, end-of-life fluence values are not expected to increase beyond those already considered and, consequently, the adjusted reference temperature used as the basis for the Technical Specification pressure/temperature curves is also expected to be bounding. The surveillance capsule withdrawal schedule and the number of capsules do not need to be changed for uprate for the same reason.

3.3.2 Reactor Internals and Pressure Differentials

The reactor internal component evaluation for power uprate is governed by load combinations that include reactor internal pressure difference (RIPD), LOCA, and seismic loads. The evaluation of the reactor internals uses ASME Boiler and Pressure Vessel Code Section 3, Subsection NB, as the basis for the design acceptance criteria. The specific applicable Code Edition for the reactor pressure vessel, including the shroud support, is the 1965 Edition with Addenda to and including Summer 1967.

3.3.3 Reactor Vessel Integrity

The feedwater nozzle cracking identified in NUREG-0619 is primarily associated with startup and shutdown cycles, which are not significantly affected by power uprate. High cycle fatigue during final feedwater temperature reduction operation has been evaluated and determined to be acceptable at uprated conditions.

3.3.3.1 Design Conditions

3.3.3.2 Normal and Upset Conditions

3.3.3.3 Emergency and Faulted Conditions

3.3.4 Steam Separator/Dryer

An evaluation was performed to determine the impact of power uprate on steam separator/dryer performance for Brunswick. The evaluation concluded that separator and dryer performance are acceptable for power uprate conditions. For an event such as loss of feedwater flow, the evaluation indicated that the moisture content of the steam at the turbine inlet may increase for a short duration; however, this is not a safety issue but an operational issue related to turbine life.

3.4 Reactor Recirculation System (RRS)

3.5 Plant Piping Systems

The effects of power uprate have been evaluated for the following Nuclear Steam Supply system (NSSS) and balance-of-plant (BOP) piping systems:

- Main steam
- Recirculation loop
- Condensate and feedwater
- Extraction Steam
- Heater vents and drains
- CRD insert and withdraw
- Reactor pressure vessel (RPV) head vent line
- Reactor Core Isolation Cooling (RCIC)
- Residual Heat Removal (RHR)
- High Pressure Coolant Injection (HPCI)
- Reactor Water Cleanup (RWCU)
- Core Spray System
- Standby Liquid Control System

Section 3.2 demonstrates that the RCPB piping will remain below the Code pressure limit during the most severe pressurization transient.

3.5.1 Evaluation Method and Approach

3.6 Main Steamline Flow Restrictors

The main steamline flow restrictor design bases are:

- The main steamline flow restrictor is designed to limit the loss of coolant and the release of radioactive materials from the reactor vessel, following a steamline rupture outside of the primary containment, to the extent that the reactor vessel water level does not fall below the top of the core within the time required to close the main steamline isolation valves.

- The main steamline flow restrictor is designed to withstand the maximum pressure differential expected across the restrictor following complete severance of a main steamline.

The effects of power uprate on the main steamline flow restrictor design bases have been evaluated and found to be acceptable for the following reasons:

3.7 Main Steam Isolation Valves (MSIV)

The main steam isolation valves (MSIVs) are designed to limit the loss of reactor coolant in case of a major leak from the steam piping outside the primary containment, and to limit the release of radioactive material in case of either a gross release of radioactive material into the reactor coolant or a major leak from the nuclear system inside the primary containment.

The MSIVs have been evaluated. The MSIV operating conditions under power uprate remain within the MSIV design conditions. The Brunswick Units 1 and 2 evaluation results are consistent with the bases and conclusions in Section 4.7 of Reference 1.

3.8 Reactor Core Isolation Cooling (RCIC) System

The Reactor Core Isolation Cooling (RCIC) System is designed to maintain the reactor vessel water level above Low Level 1 in the event of a transient which results in the loss of all feedwater flow or reactor isolation, until the reactor is depressurized to a level where the

Residual Heat Removal (RHR) System can be placed in operation. The RCIC System has been evaluated at the uprated power conditions and was found to meet the reactor Low Level 1 design criteria at its current design rated flow of 400 gpm. The system was also found to have the capability to deliver its design rated flow at the increased reactor pressure resulting from the SRV safety function pressure setpoint increase of 25 psi and the allowable setpoint tolerance of 3%. Power uprate will not have any effect on the availability or the reliability of the RCIC System. A RCIC System flow test is being included in the startup test program for each unit. The evaluation results are consistent with the generic evaluation in Section 4.2 of Reference 1.

The RCIC System startup response is dependent on the reactor pressure. The increase in steam pressure will slightly increase the initial acceleration of the turbine and pump rotors. The existing turbine startup characteristics have been reviewed, and the turbine control system will be able to accommodate the increase in acceleration rate without increasing the risk of a turbine overspeed trip.

3.9 Residual Heat Removal (RHR) System

The Residual Heat Removal (RHR) System is designed to restore and maintain the coolant inventory in the reactor vessel and provide primary system decay heat removal following reactor shutdown for both normal and post accident conditions. The RHR System is designed to operate in the Low Pressure Coolant Injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The effects of power uprate on the LPCI mode are discussed in Subsection 4.2.2. The effects of power uprate on the remaining modes are discussed in the following paragraphs.

3.9.1 Shutdown Cooling Mode

The Brunswick units are not committed to Regulatory Guide (R.G.) 1.139, which requires demonstration of cold shutdown (temperature of 212°F) within 36 hours assuming the most limiting single failure. A R.G. 1.139 evaluation has not been performed for Brunswick; however, based on the results of evaluations for other plants and the results of the Brunswick evaluation discussed above (temperature of 125°F is reached within 21 hours), the 36-hour criterion is expected to be met with substantial margin.

3.9.2 Suppression Pool Cooling (SPC) and Containment Spray (CC) Modes

The SPC and CC modes are designed to provide sufficient cooling to maintain the containment and suppression pool temperatures and pressures within design limits following design basis LOCA conditions.

The effect of power uprate on peak suppression pool temperature after a design basis LOCA is described in Section 4.1.1. The analysis confirms that the pool temperature stays below the design limit, and the capability of this mode is acceptable for power uprate.

Power uprate increases the containment spray temperature by approximately 3°F. This increase has a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure, since these parameters reach peak values prior to actuation of the containment spray. The capability of the CC mode is therefore acceptable for the power uprate.

3.9.3 Low Pressure Coolant Injection (LPCI) Mode

The LPCI mode is discussed in Subsection 4.2.2.

3.9.4 Fuel Pool Cooling Assist Mode

During normal plant shutdown, with the vessel head removed, the RHR System can be aligned to assist the Fuel Pool Cooling and Cleanup System to maintain the fuel pool temperature within acceptable limits (Table 6-2). The analysis in Section 6.3 indicates that the fuel pool temperature will remain within limits under power uprate conditions, and therefore the capability of the Fuel Pool Cooling Assist mode is acceptable for the power uprate.

3.10 Reactor Water Cleanup System (RWCU)

Operation of the plant at the uprated power level slightly increases the temperature and pressure within the Reactor Water Cleanup (RWCU) System. This 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. The system is capable of performing this function at the uprated power level.

3.11 Main Steam and Feedwater Piping

The main steam and feedwater piping within the RCPB is addressed in Section 3.5.

3.12 Balance-of-Plant (BOP) Piping

The adequacy of the BOP piping design for operation at the uprated conditions. All piping except the systems is addressed in Section 3.5.

3.12.1 Piping System Evaluation

3.12.2 Pipe Support Evaluation

Operation at the uprated 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 power uprate, such as seismic and deadweight, the overall combined support load increase is insignificant.

The supports of the systems with the most loading increases (main steam and feedwater systems) were reviewed to determine if there is sufficient margin to accommodate the increased loadings. This review shows that there is adequate difference between the original design stresses and code limits of the supports to accommodate the load increase within the appropriate code criteria. The existing conservatisms, of the existing calculation, are introduced by the use of the lowest code allowables for various plant loading conditions, the use of generic enveloping design loads instead of actual loads, and the conservative load application on base plates, anchor bolts, and lugs.

3.12.3 Erosion/Corrosion

Erosion/corrosion may be affected by the increased flow rates, higher operating temperatures, and change in moisture content of two-phase streams; however, the differences due to power uprate are small and should not cause a new erosion/corrosion concern. Brunswick has a formal erosion/corrosion program for monitoring pipe wall thickness. Any increase in the rate of thinning that occurs as the result of power uprate operation will be identified through this program.

3.13 References

1. GE Nuclear Energy, *Generic Evaluations of General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-proprietary), March 1992; and Supplements.

2. GE Nuclear Energy, *General Electric Standard Application for Reactor Fuel - GESTAR II, United States Supplement,* NEDE-24011-P-A-10-US, March 1991.

3. GE Nuclear Energy, *Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDO-31897, Class I (Non-proprietary), February 1992; NEDC-31897P-A, Class III (Proprietary), May 1992.

Table 3-1
RIPDs FOR NORMAL CONDITIONS (psid)[1]

Parameter	Pre-Uprate [2]	Power Uprate
Core Plate & Guide Tube	26.0	26.9
Shroud Support Ring & Lower Shroud	33.0	34.7
Shroud	7.0	7.8
Shroud Head	7.2	7.9
Top Guide	0.7	0.8
Steam Dryer	0.3	0.4

[1] Values shown are for Unit 2 and bound those for Unit 1.

[2] 100% power/100% flow.

Table 3-2

RIPDs FOR UPSET CONDITIONS (psid) [1]

Parameter	Pre-Uprate [2]	Power Uprate
Core Plate & Guide Tube	28.4	29.3
Shroud Support Ring & Lower Shroud	35.4	37.1
Shroud	10.5	11.7
Shroud Head	10.8	11.8
Top Guide	<1.0	1.1
Steam Dryer	0.4	0.6

[1] Values shown are for Unit 2 and bound those for Unit 1.

[2] 104% power/100% flow.

Table 3-3
RIPDs FOR FAULTED CONDITIONS (psid)[1]

Parameter	Pre-Uprate[2]	Power Uprate
Core Plate & Guide Tube	34.0	34.0
Shroud Support Ring & Lower Shroud	55.0	55.0
Shroud	29.0	30.0
Shroud Head	29.0	30.0
Top Guide	4.0	4.0
Steam Dryer	10.0	10.0

(1) Values shown are for Unit 2 and bound those for Unit 1.

(2) 104% power/100% flow.

Table 3-4

SUMMARY OF MAXIMUM STRESSES AND LOCATIONS FOR REACTOR INTERNALS AT 105% POWER UPRATE

A. Upset Conditions

Max Stress Components		Max. Stress Comparison (psi)	
Component	Location	Uprated	Allowable (psi)
Shroud Repair Bracket	Bolt connection holes	10,410	20,930
Shroud	Shroud-to-shroud support weld	3,927	7,640
Shroud Head Assembly	Bolts	28,619	34,950
Shroud Support	Vessel shell attachment	<26,650	26,700

B. Faulted Conditions

Max Stress Components		Max. Stress Comparison (psi)	
Component	Location	Uprated	Allowable (psi)
Shroud Repair Bracket	Bolt connection holes	29,240	41,850
Shroud	Shroud-to-shroud support weld	14,310	15,280
Shroud Head Assembly	Bolts	60,319	69,900
Shroud Support	Stilt stub	30,540	35,000

Table 3-5
FATIGUE USAGE FACTORS OF LIMITING COMPONENTS

Component	Fatigue Usage Factor	
	At Current Power	At Uprated Power
Feedwater Nozzle - Unit 1	0.747	0.856
Feedwater Nozzle - Unit 2	0.73	0.96
Closure Region Bolts - Both Units	0.8	0.81
Recirculation Inlet Nozzle - Both Units	0.81	0.86
Core Spray Nozzle - Both Units	0.98	0.96*

* The fatigue usage factor for the core spray nozzle decreased relative to the evaluation at current power due to the removal of unnecessarily conservative assumptions. An evaluation at the current power level using the same analysis basis as used for the uprated power is expected to show the same trend as the other results in this table.

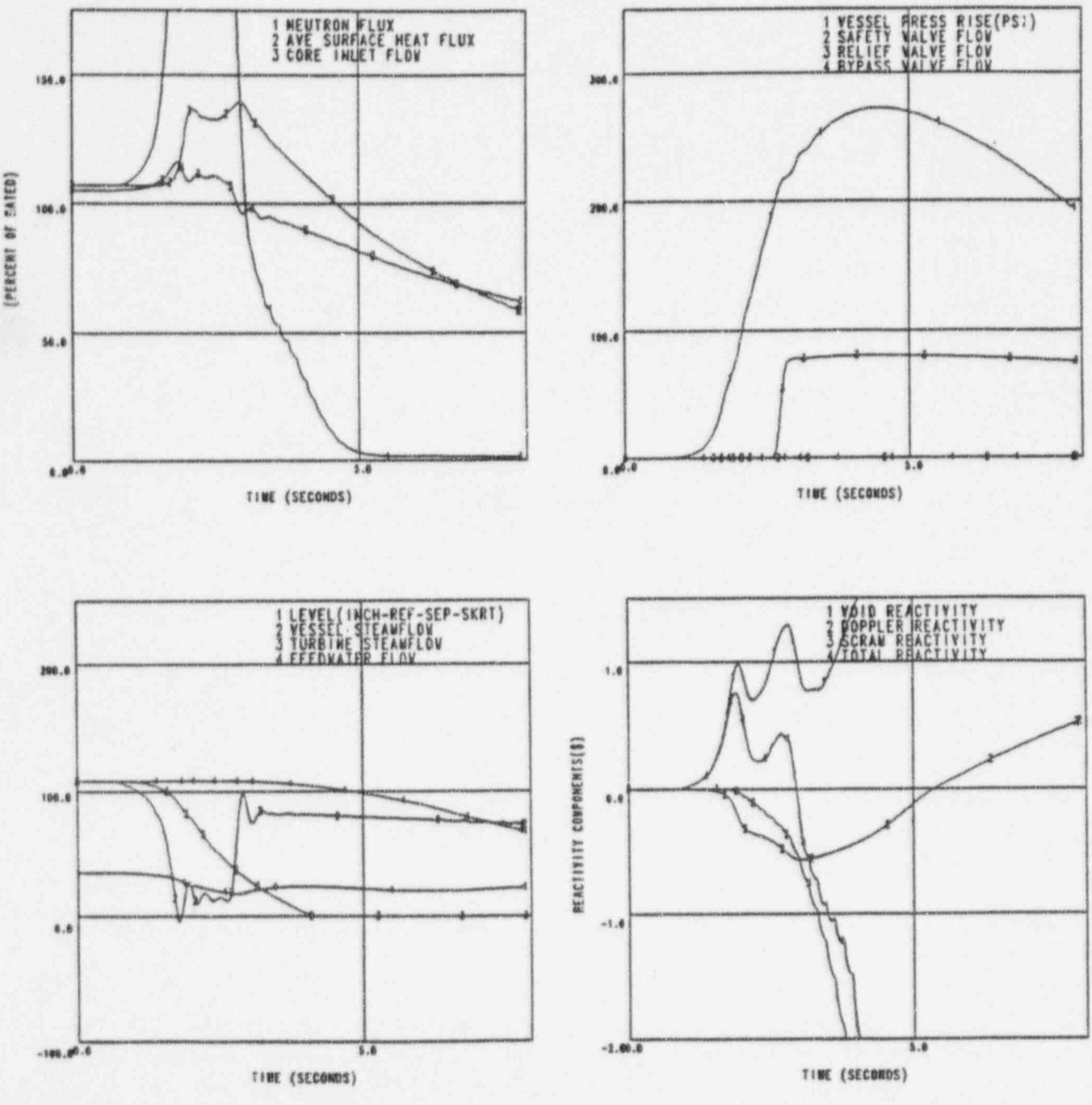

Figure 3-1. Response to MSIV Closure With Flux Scram (102% of Uprated Power, 105% Core Flow and 1060 psia Initial Dome Pressure)

4.0 ENGINEERED SAFETY FEATURES

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 6, that applies to power uprate.

4.1 Containment System Performance

4.1.1 Containment Pressure and Temperature Response

Figures 4-1 and 4-2 provide the short-term drywell and wetwell pressure and temperature responses to the DBA-LOCA. Figure 4-3 provides the long-term suppression pool temperature response.

4.1.1.1 Long-Term Suppression Pool Temperature Response

Bulk Pool Temperature:

Local Pool Temperature With SRV Discharge:

The local pool temperature limit for SRV discharge is specified in NUREG-0783, because of concerns resulting from unstable condensation observed at high pool temperatures in plants without quenchers. Reference 11 provides justification for elimination of this limit for plants with quenchers on the SRV discharge lines. Because this plant has quenchers, no evaluation of this limit is necessary, as discussed in Supplement 1 of Reference 12; however, the local suppression pool temperature has been evaluated for power uprate and found to be acceptable relative to the NUREG-0783 limits.

4.1.1.2 Containment Gas Temperature Response

4.1.1.3 Short-Term Containment Pressure Response

4.1.2 Containment Dynamic Loads

4.1.2.1 LOCA Containment Dynamic Loads

4.1.2.2 Safety/Relief Valve (SRV) Containment Dynamic Loads

4.1.2.3 Subcompartment Pressurization

4.1.3 Containment Isolation

4.2 Emergency Core Cooling Systems (ECCS)

Each ECCS is discussed in the following subsections. The effect of each system's functional capability, due to power uprate and the increase in RPV dome pressure, is addressed. The ECCS performance evaluation is contained in Section 4.3.

The NPSH requirement for the Brunswick units is based on zero containment pressure and peak bulk temperature determined as described in Section 6.3.2.2.5 of the UFSAR. The design basis event for determining NPSH margin was evaluated, and a peak suppression pool temperature of 189°F was calculated. Based on this temperature and with no credit for containment pressure, it was found that there is approximately 20.4 ft of NPSH available to the Residual Heat Removal (RHR) pumps, and 16.6 ft of NPSH available to the Core Spray (CS) pumps. This provides a margin to the required NPSH of 5.4 ft for the RHR pumps and 2.6 ft for the CS pumps. Therefore, adequate NPSH is available to the RHR and CS pumps.

4.2.1 High Pressure Coolant Injection (HPCI)

The HPCI System is designed to provide sufficient core cooling to prevent excessive fuel cladding temperature in the event of a hypothetical small break loss-of-coolant accident that does not depressurize the reactor quickly enough to permit timely operation of the low pressure

ECCS. The HPCI System also serves as backup to the RCIC System. The Brunswick units have implemented the guidance of GE Services Information Letter 480. The uprated power conditions were found to satisfy the core cooling assumptions of the SAFER/GESTR LOCA analysis (Reference 13) at its current design rated flow of 4250 gpm. The system was also found to have the capability to deliver its design rated flow at the increased reactor pressure resulting from the SRV safety function pressure setpoint increase of 25 psi and the allowable setpoint tolerance of 3%. Power uprate will not have any effect on the availability or the reliability of the HPCI System. A HPCI System flow test is being included in the startup test program for each unit. The evaluation results are consistent with the generic evaluation in Section 4.2 of Reference 12.

4.2.2 RHR System (Low Pressure Coolant Injection)

The adequacy of the Low Pressure Coolant Injection (LPCI) mode of the RHR System and the other ECCS to provide reactor core cooling during a LOCA is addressed in Section 4.3. The hardware capability of the equipment associated with the LPCI mode is bounded by the generic evaluation provided in Section 4.1 of Reference 12.

4.2.3 Core Spray (CS) System

The hardware capability of the Core Spray System is bounded by the generic evaluation in Section 4.1 of Reference 12. The adequacy of the CS System and the other ECCS to provide reactor core cooling during a LOCA is addressed in Section 4.3.

4.2.4 Automatic Depressurization System (ADS)

The ADS uses safety/relief valves to reduce reactor pressure following a small break LOCA with high pressure ECCS failure. This function allows LPCI and low pressure CS to flow to the vessel. The ADS initiation logic and ADS valve control are adequate for power uprate conditions. ECCS design assumes a minimum flow capacity for the SRVs, and that ADS initiate (after a time delay) on low water level and a signal that one CS pump or two RHR pumps are running with adequate discharge pressure. The flow capacity and ability to initiate ADS on appropriate signals are not affected by power uprate.

4.3 ECCS Performance Evaluation

The Emergency Core Cooling Systems (ECCS) are designed to provide protection against hypothetical LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and their analysis models satisfy the requirements of 10CFR50.46 and 10CFR50 Appendix K. The results of the ECCS-LOCA analysis using NRC approved methods (SAFER/GESTR) are provided in Reference 13. The evaluations were performed at 110% power (nominal case) and 102% of 110% of power (Appendix K case), and therefore bound the 105% power uprate. The SAFER/GESTR code is Brunswick's current licensing basis (prior to power uprate), and therefore an update for a previous ECCS code is not a part of the Brunswick power uprate license amendment.

The single loop operation (SLO) option is valid for power uprate. SLO is limited to 88% of the current power level. At uprated power conditions, this corresponds to 83.8% of the uprated power level.

4.4 Standby Gas Treatment System (SGTS)

The standby gas treatment system is designed to minimize offsite dose rates during venting and purging of both the primary and secondary containment atmosphere under accident or abnormal conditions, by removing fission products, particularly halogens, from the vented steam through a high efficiency filtration system. The capacity of the SGTS was selected to provide one secondary containment air volume change per day and thereby maintain the reactor building at a slight negative pressure. This capability is not impacted by the uprate.

The charcoal filter beds are unaffected by uprate. The total post-LOCA iodine loading increases slightly at the uprated conditions, but is well below the original design capability of the filters.

4.5 Other ESF Systems

Per Regulatory Guide 1.70, some of the following systems are not ESF systems, but are being addressed here for completeness.

4.5.1 Post-LOCA Combustible Gas Control

The Combustible Gas Control (CGC) System at Brunswick consists primarily of a Containment Atmosphere Dilution (CAD) System which is capable of maintaining the containment atmosphere below 5 volume% O_2. Power uprate will cause an increase of radiolytic oxygen proportional to the increase in plant thermal power. Following a LOCA, the CAD System will be initiated earlier, and the limiting pressure of one-half of the design pressure will be reached earlier. The initiating time will be well beyond the regulatory minimum of 30 minutes and the limiting pressure will be reached beyond 30 days.

4.5.2 Emergency Cooling Water System

Safety-related and nonsafety-related water systems are addressed in Section 6.4.

4.5.3 Emergency Core Cooling Auxiliary Systems

Power dependent HVAC and other auxiliary systems are addressed in Sections 6.6 and 6.8.

4.5.4 Main Control Room Atmosphere Control System

This system is not affected by power uprate. The impact of power uprate on the whole body and thyroid dose in the main control room resulting from a LOCA is summarized in Table 9-3.

4.5.5 Standby Power System

Standby power is addressed in Sections 6.1 and 6.2.

4.6 References

1. GE Nuclear Energy, *Generic Guidelines For GE Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDO-31897, Class I (Non-proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.

2. NEDM-10320, *The GE Pressure Suppression Containment System Analytical Model,* March 1971; Supplement 1, May 1971; Supplement 2, June 1973.

3. NEDO-20533, *The General Electric Mark III Pressure Suppression Containment System Analytical Model,* June 1974.

4. NEDO-21052, *Maximum Discharge of Liquid-Vapor Mixtures from Vessels,* General Electric Company, September 1975.

5. NEDE-20566-P-A, *General Electric Model for LOCA Analysis in Accordance With 10CFR50 Appendix K,* September 1986.

6. NUREG-0800, U.S. Nuclear Regulatory Commission, Standard Review Plan, Section 6.2.1.1.C, *Pressure - Suppression Type BWR Containments,* Revision 6, August 1984.

7. NUREG-0661, *Mark I Containment Long-Term Program Safety Evaluation Report,* July 1980.

8. NEDO-21888, *Mark I Containment Program Load Definition Report,* Revision 2, November 1981.

9. United Engineers and Constructors, Inc., *Plant Unique Analysis Report, Mark I Containment Program, Brunswick Steam Electric Plant, Units 1 and 2,* October 1982.

10. Letter, A. C. Thadani (NRC) to G. Sozzi (GE), "Use of SHEX Computer Program and ANSI/ANS 5.1 1979 Decay Heat Source Term for Containment Long-term Pressure and Temperature Analysis," July 13, 1993.

11. NEDO-30832, *Elimination of Limit on Local Suppression Pool Temperature for SRV Discharge with Quenchers,* Class I, December 1984.

12. GE Nuclear Energy, *Generic Evaluations of General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDC-31984-P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-proprietary), March 1992; and Supplements.

13. GE Nuclear Energy, *Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (Revision 2),* NEDC-31642P, Class III (Proprietary), July 1990.

Table 4-1
CONTAINMENT PERFORMANCE RESULTS

Parameter	Current Rated Power	Uprated Power	Limit
Peak Drywell Pressure (psig)	49.4 (UFSAR) 36.8*	38.1*	62
Peak Bulk Pool Temperature (°F)	205 (UFSAR) 197 (current method)	201	220
Drywell Temperature (°F)	283*	284	340**

* Mark I LTP method.

** The drywell temperature design limit is reported in the UFSAR as 281°F. However, drywell liner stresses and equipment in the drywell can actually withstand temperatures of up to 340°F.

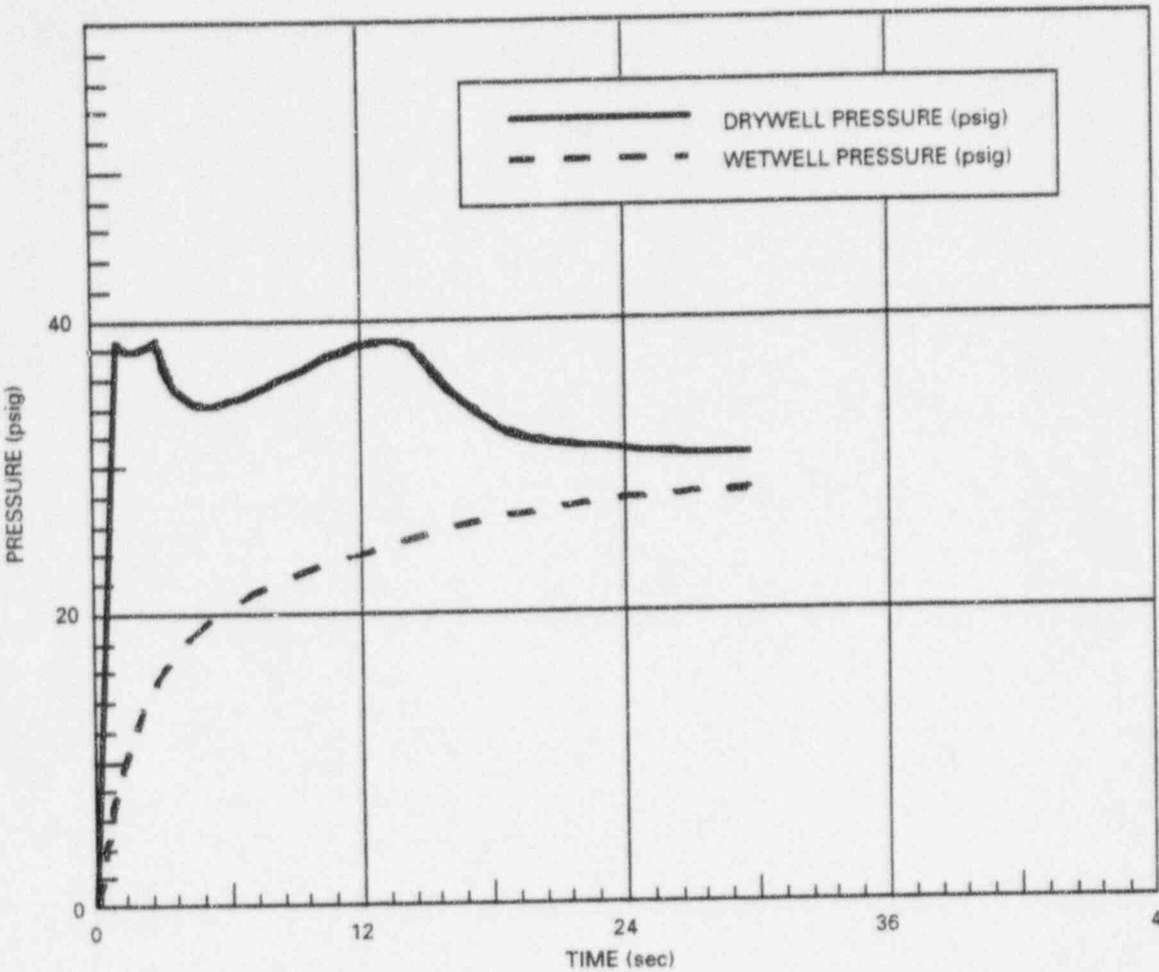

Figure 4-1. DBA-LOCA Short-Term Drywell and Wetwell Pressure Response (102% of Uprated Power)

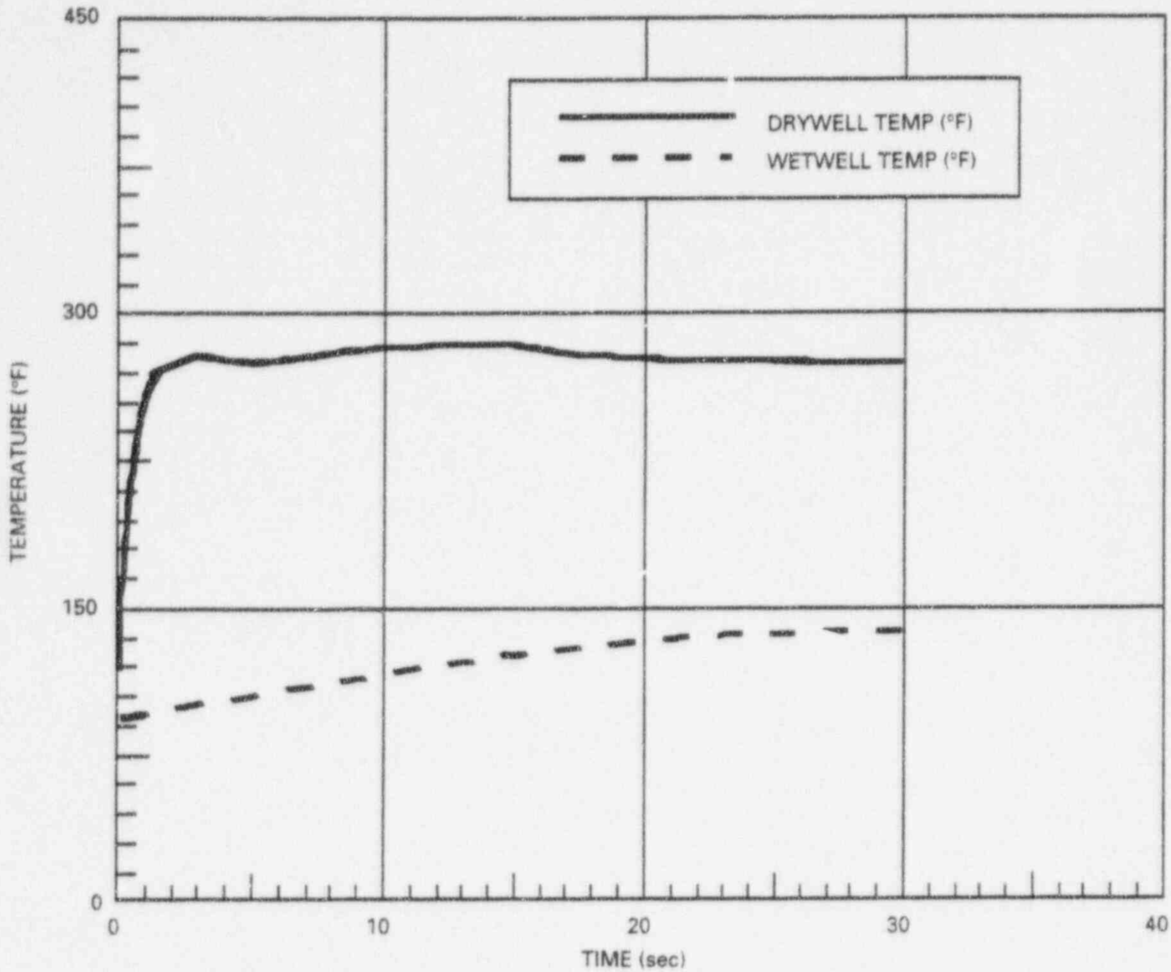

Figure 4-2. DBA-LOCA Short-Term Drywell and Wetwell Temperature Response (102% of Uprated Power)

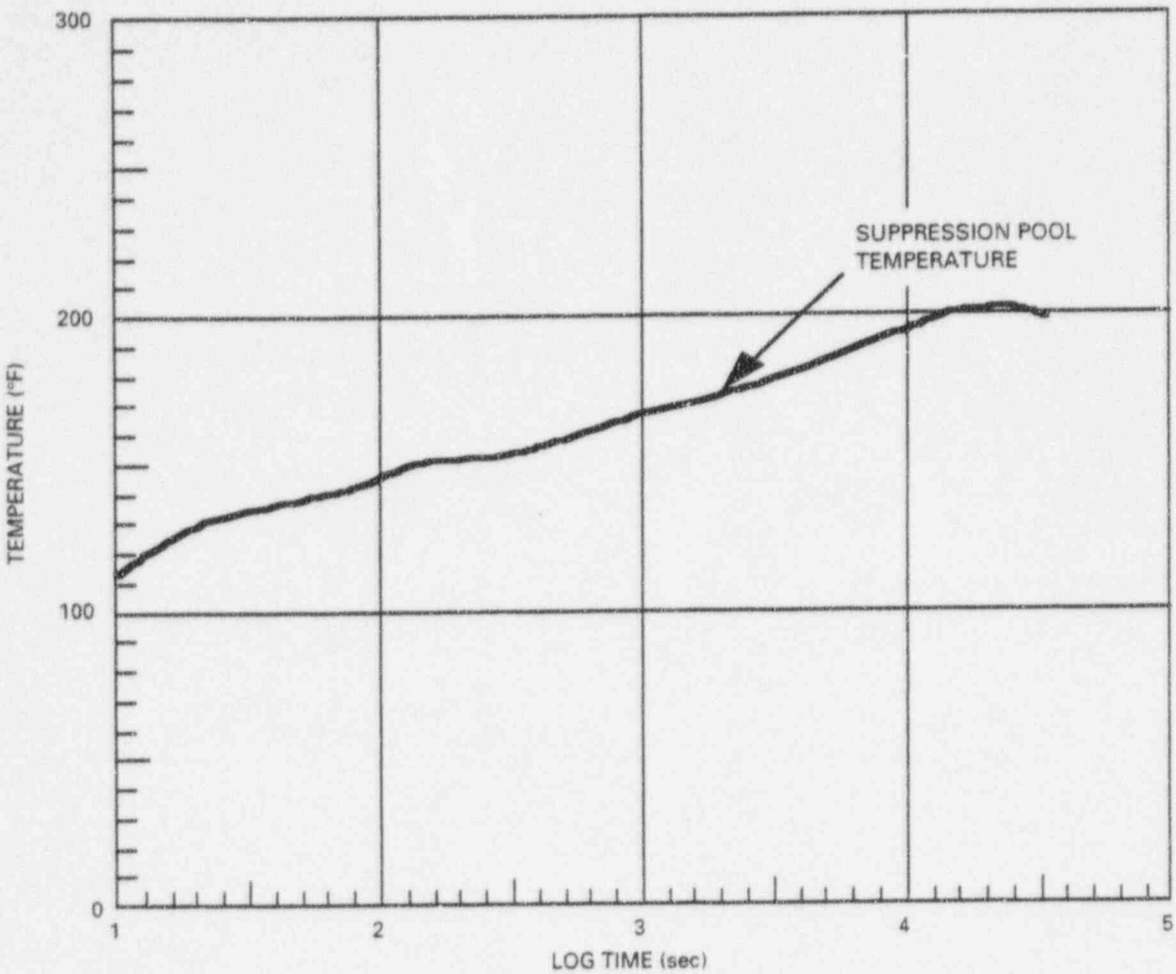

Figure 4-3. Long-Term DBA-LOCA Suppression Pool Temperature Response (102% of Uprate Power)

5.0 INSTRUMENTATION AND CONTROL

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 7, as it applies to power uprate.

Indications in the control room will be modified as necessary to implement power uprate. This means that the uprated operating parameters will be indicated as 100% of the new license conditions wherever applicable.

5.1 Nuclear Steam Supply System

Plant process variables, instrument setpoints and Regulatory Guide 1.97 instrumentation that could be affected by power uprate were evaluated. The following summarizes the results of those evaluations.

5.1.1 Neutron Monitoring System

The average power range monitor (APRM) power signals will be rescaled to the uprated power, and the fixed neutron flux APRM trip setpoints will be unchanged. The rescaling of the APRM system automatically adjusts the trip values without changing the actual setpoints. The flow-biased APRM rod block and simulated thermal power monitor (STPM) scram setpoints are modified (see Section 5.1.2) to maintain the same setpoint margin between the rod block warning and scram protection.

The current power dependent Rod Block Monitor (RBM) setpoints at the uprated power level were assumed in the Rod Withdrawal Error (RWE) transient analysis (results are in Table 9-2). That analysis demonstrates that RBM performance at the uprated power conditions is adequate to ensure that local power excursions due to any potential RWE are maintained within acceptable limits.

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

The neutronic life of the LPRM detectors and radiation level of the traversing in-core probe (TIP) may be affected slightly due to the higher power level. The effect is proportional to the power uprate.

5.1.2 Instrument Setpoints

The determination of instrument setpoints is based on conservative licensing analyses and setpoint methodology. The settings are selected with sufficient margin to preclude inadvertent initiation of the protective action while assuring that adequate difference is maintained between the system settings and the actual limit.

As a result of the increase in the reactor power and dome pressure, all potentially impacted analytical limits for setpoints were assessed. This section provides those analytical limits affected by the power uprate. Setpoints will be recalculated to provide adequate allowances between the operational settings and the analytical limits provided in Table 5-1, to ensure the necessary safety functions. CP&L setpoint calculations are performed in accordance with the methodology of ISA-S67.04, Parts I and II, September 1994. This is the latest approved version of the standard methodology endorsed by Regulatory Guide 1.105, Revision 2.

Table 5-1 provides a summary of the impacted analytical limits for setpoints resulting from the power uprate safety analyses.

5.1.2.1 RPV High-Pressure Scram

During a pressure increase transient not terminated by direct scram or high flux scram, the high-pressure scram will terminate the transient. The reactor vessel high-pressure scram signal settings are maintained slightly above the reactor vessel maximum normal operating pressure and below the specified analytical limit. The setting permits normal operation without spurious scram, yet provides adequate difference to the maximum allowable reactor vessel pressure. As a result of the pressure increase associated with uprate, the analyzed scram setpoint on reactor high pressure will be increased as shown in Table 5-1. The nominal setpoint value will be increased by a similar amount to avoid spurious scrams and yet provide adequate difference to the maximum allowable pressure.

5.1.2.2 High-Pressure RPT

The anticipated transient without scram recirculation pump trip (ATWS-RPT) is provided to trip the pumps during plant transients associated with increases in reactor vessel dome pressure and low reactor water level. The ATWS-RPT is designed to provide negative reactivity by reducing core flow during the initial part of an ATWS.

The major consideration for the pressure setpoint is the increase in peak pressure during a hypothesized ATWS event because of the higher initial power value. The effects of increasing safety/relief valve and dome pressure setpoints have been evaluated in Sections 3.1, 3.2, 9.1 and 9.3.1, using the analytical limits shown in Table 5-1. These evaluations found a resulting increase in the peak vessel bottom pressure; however, the calculated peak pressure will remain below the 1500 psig limit for ATWS. Therefore, the current ATWS-RPT analytical limit is acceptable for power uprate.

5.1.2.3 Safety/Relief Valve

As a result of the increase in reactor operating dome pressure due to power uprate, the SRV analytical limits for setpoints are revised (see Table 5-1) and will continue to ensure appropriate pressure relief. These values were used in the overpressure protection and transient analyses discussed in Sections 3.2 and 9.1.

5.1.2.4 Neutron Monitoring System

The APRM flow-biased rod block and simulated thermal power (STP) trip setpoints will be changed to account for the lowering of the zero-flow intercept power level and the raising of the rated power intercept flow level, as shown in Figure 2-1.

5.1.2.5 Main Steam High Flow Isolation

The analytical limit for uprate remains at 140% of the uprated steam flow. The instrumentation will be recalibrated for the increase in steam flow shown in Table 1-2. This ensures that sufficient difference to the trip setpoint exists to allow for normal plant testing of the MSIVs and turbine stop valves.

5.1.2.6 Main Steamline High Radiation Scram

The main steamline high radiation scram function has been removed from Unit 1, and is scheduled for removal from Unit 2 prior to implementation of power uprate.

5.1.2.7 Turbine Stop Valve Closure and Turbine Control Valve Fast Closure Scram Bypass

The first stage turbine pressure setpoint is chosen to allow operational margin so that scrams can be avoided, by transferring steam to the turbine bypass system during turbine/generator (T/G) trips at low power.

The transient analysis for uprate was performed with the analytical limit raised according to the current basis (e.g., equivalent to 30% power). As discussed in Section F.4.2, paragraph (c) of Reference 1, this approach maintains the safety basis for the setpoint, and the small absolute increase in the effective setpoint does not make any significant difference in the transient analysis results.

5.1.2.8 Rod Worth Minimizer (RWM) Low Power Interlocks

The RWM Low Power Interlocks provide signals to ensure RWM operability at thermal powers ≤10% of rated. Allowing the interlocks to be operable up to 10% of uprated power is in the conservative direction, and thus, no change in the operability range is needed.

5.1.2.9 Reactor Water Level Instruments

The water level instruments are to be re-calibrated for power uprate conditions. The indicated water level alarms and trips will not change. Power uprate will not adversely affect the reactor vessel water level instrumentation purging system; the system's flow rate is checked periodically and will be adjusted if necessary to account for the increase in reactor pressure due to power uprate.

5.1.2.10 Main Steamline Closure on High Tunnel Temperature

The normal operating temperature is expected to increase only 2-3°F with power uprate, and therefore the current setpoint provides adequate margin. The temperature will be monitored during uprate operation.

5.2 Balance-of-Plant (BOP) – Power Conversion and Auxiliary Systems

5.2.1 Control Systems Evaluation

Operation of the plant at the uprated power level has minimal impact on the BOP system instrumentation and control devices. Based on uprated operating conditions for the power conversion and auxiliary systems, the process control valves and instrumentation have sufficient range/adjustment capability for use at the expected uprated conditions. No safety-related BOP system setpoint change is required as a result of the uprate.

5.2.1.1 Pressure Control System

The objective of the pressure control system (PCS) is to provide a fast and stable response to pressure and steam flow disturbances so that the reactor pressure is controlled within its allowed high and low limits. The PCS consists of the pressure regulation system, turbine control valve system, and steam bypass valve system. At uprated power conditions, the operating turbine inlet pressure becomes an important consideration in obtaining a desirable operating point on the turbine control valves. Adequate control valve range must be available to assure the ability of the pressure control system to respond to system disturbances demanding increased steam flow for the purpose of minimizing pressure excursions. Sufficient control pressure range during system disturbances is available with power uprate. Thus, the PCS is adequate for power uprate conditions.

5.2.1.2 EHC Turbine Control System

No modification to the turbine control valves or the turbine bypass valves are required for operation at the uprated throttle pressure conditions. Normal manual operator setpoint control is used to establish the new pressure setpoint using modified operating procedures. No control stability problems associated with the increase in steam pressure are expected for power uprate operation.

5.2.1.3 Pressure Regulator

The pressure regulator setpoint is increased to achieve proper control characteristics for the turbine control valve for the power uprate. The reactor operator adjusts the pressure regulator setpoint pressure to maintain the turbine inlet pressure within its required operating range.

5.3 References

1. GE Nuclear Energy, *Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDC-31897, Class I (Non-proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.

Table 5-1

ANALYTICAL LIMITS FOR SETPOINTS

Parameter	Analytical Limits	
	Current	**Power Uprate**
APRM Basis	Calibrated to 2436 MWt (rated power)	Calibrated to 2558 MWt (rated power)
APRM Simulated Thermal Power Scram Clamped[1]	117%	117%
APRM Neutron Flux Scram	121%	121%
Vessel High Pressure Scram	1086 psia	1111 psia
High Pressure ATWS RPT	1135 psia	1135 psia
Safety/Relief Valve Settings (psig)[2]	1116 1126 1136	1164 1174 1185
Turbine First-Stage Scram Bypass Pressure (%Power)	30.0%	30.0%
Main Steam High Flow	140%	140%
Main Steam High Flow (Unit 2 only Modes 2 and 3)	40%	40%

NOTE:

(1) Analytical Basis – No credit taken for flow bias.

(2) Includes tolerances.

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapters 8 and 9, that applies to power uprate.

6.1 AC Power

Plant electrical characteristics are given in Table 6-1.

6.1.1 Offsite Power System

Conformance to General Design Criteria 17 (10CFR50, Appendix A), which addresses onsite and offsite electrical supply and distribution systems for safety-related components, is not affected. No modifications are required to the electric power system because of power uprate; however, existing analyses (e.g., electrical system loads) will be revised to incorporate the actual increases in motor power due to the increased pumping loads in the condensate, feedwater, and recirculation systems.

The isolated phase bus duct is adequate for both rated voltage and low voltage current output.

The main transformers and the associated switchyard components (rated for maximum transformer output) are adequate for the uprated transformer output.

6.1.2 Onsite Power Distribution System

The onsite power distribution system consists of transformers, numerous buses, and switchgear that support or are powered by the buses and switchgear.

Station loads under normal operation/distribution conditions are computed based on necessary equipment loads. Existing analyses of electrical loads will be reviewed, and revised if necessary to incorporate the higher loads needed by some of the unit's motors. Operation at the uprated level will be achieved by utilizing existing equipment operating at or below the nameplate rating.

Station loads under emergency operation/distribution conditions (Emergency Diesel Generators) are based on needed brake horsepower, except for core spray and RHR pumps where a conservatively high flow brake horse power (BHP) is used. Operation at the uprated 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 4.16 kVAC powered ECCS equipment for power uprate, and only a small increase (approximately proportional to the increase in reactor pressure) is needed for the 480 VAC Standby Liquid Control System motors. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) will not be increased with power uprate, and the as-designed emergency power system will remain adequate.

6.2 DC Power

The DC loading requirements in the USAR were reviewed, and no power dependent loads were identified.

The DC power distribution system provides control and motive power for various systems/components within the plant. Operation at the uprated level will not increase any loads beyond nameplate rating or revise any control logic; therefore the existing DC power distribution system is adequate for uprated conditions.

6.3 Fuel Pool Cooling

The fuel pool cooling system consists of two fuel pool cooling pumps, two heat exchangers, two filter demineralizers, two skimmer surge tanks, and associated piping, valves and instrumentation. During normal plant operation (non-outage), one pump and one heat exchanger are sufficient to maintain pool temperature within design limits. During refueling, two pumps and two heat exchangers are utilized.

As a result of operation at the uprated power level, the spent fuel pool heat load will slightly increase as shown in Table 6-2.

The fuel pool cooling and cleanup adequacy is determined by calculating the heat load generated by a full core discharge plus remaining spaces filled with used fuel discharged at regular intervals, and by calculating the bulk pool temperature.

Each reload will affect the decay heat generation in the spent fuel discharged from the reactor. This evaluation considered the expected heat load in the spent fuel storage pool at the uprated conditions, and confirms the capability of the fuel pool cooling system to maintain adequate fuel pool cooling. The results are summarized in Table 6-2.

The results of an expected refueling cycle analysis show that the maximum heat load in the spent fuel storage pool will be less than the heat removal capability of the two fuel pool cooling heat exchangers. Therefore, it is concluded that power uprate will not have any negative effect on the capability to keep the fuel pool temperature at or below the design temperature and maintain adequate fuel pool cooling for normal discharge conditions.

Similarly, full core discharge heat load in the fuel storage pool is expected to reach a maximum immediately after the full core discharge. If the heat load is higher than the total design capacity of the fuel pool heat exchangers, the RHR system is more than adequate to provide the required additional spent fuel pool cooling capacity until the heat load diminishes enough to be accommodated by the two fuel pool heat exchangers alone. Therefore, it is concluded that the planned power uprate will not have any negative effect on the capability to maintain adequate spent fuel cooling for full core discharge conditions.

Based on the above heat load evaluations, the fuel pool equilibrium temperature is not expected to exceed system capability.

The normal radiation levels around the pool will increase by less than 5%, primarily during fuel handling operation. This increase is acceptable and will not significantly increase the operational doses to personnel or equipment.

In summary, the changes are small and are within the design limits of the affected systems and components.

6.4 Water Systems

The environmental effects of uprate will be controlled at the same level as is presently in place. That is, the plant operation will be managed such that none of the present limits (such as maximum allowed ultimate heat sink temperature) will be exceeded as a result of uprate.

6.4.1 Service Water Systems

6.4.1.1 Safety-Related Loads

The safety-related service water system is designed to provide a reliable supply of cooling water during and following a design basis accident for the following essential equipment and systems:

- RHR heat exchangers
- Emergency diesel generator coolers
- ECCS room coolers
- RHR pump seal coolers

The evaluation of the system's performance is given in the following subsections.

6.4.1.1.1 Emergency Equipment Service Water System

The safety-related performance of the service water system during and following the most demanding design basis event, the LOCA, for the following equipment and systems is not dependent upon the point of rated power operation:

- Emergency diesel generator coolers
- ECCS room coolers
- RHR pump seal coolers

The diesel generator loads and the RHR System flows remain unchanged for LOCA conditions following uprated operation. The ECCS room cooling loads also remain the same as that for rated operation because the equipment performance in these areas has remained unchanged for post-LOCA conditions. Uprate does not require the modification of the service water system.

6.4.1.1.2 Residual Heat Removal Service Water System

The containment cooling analysis in Section 4.1.1 does not assume that the post-LOCA RHR cooling capacity is increased for power uprate. Therefore, power uprate will not increase the cooling requirements on the RHR and its associated service water system.

6.4.1.2 Nonsafety-Related Loads

The service water discharge temperature results from the heat rejected to the service water system via the closed cooling water systems and other auxiliary heat loads. The major service water heat load increases from power uprate reflect an increase in main generator losses rejected to the stator water coolers, hydrogen coolers and exciter coolers in addition to increased bus cooler heat loads. The increase in service water heat loads from these sources due to uprated operation is projected to be approximately proportional to the uprate itself.

For normal operation the maximum service water heat loads occur during peak summer months. An uprated discharge temperature may be estimated assuming both realistic conditions and very conservative bounding conditions. The assumptions for each and the estimated heat load, discharge temperature and differential temperature for uprated conditions are shown in Table 6-3. These results demonstrate that the service water system is adequate for power uprate conditions.

6.4.2 Main Condenser/Circulating Water/Normal Heat Sink Performance

The main condenser, circulation water and heat sink systems are designed to remove the heat rejected to the condenser and thereby maintain adequately low condenser pressure as recommended by the turbine vendor. Maintaining adequately low condenser pressure assures the efficient operation of the turbine-generator and minimizes wear on the turbine last stage buckets.

Uprate operation increases the heat rejected to the condenser and therefore reduces the difference between the operating pressure and the required minimum condenser vacuum. The performance of the main condenser was evaluated for power uprate. This evaluation is based on a design duty over the actual yearly range of circulating water inlet temperatures, and confirms that the condenser, circulating water system and heat sink are adequate for uprated operation.

6.4.2.1 Discharge Limits

The state discharge limits (to air) were compared to the current discharges and bounding analysis discharges for power uprate, as shown in Table 6-3. This comparison demonstrates that the plant will remain within the state discharge limit, during operation at uprated power.

6.4.3 Reactor Building Closed Cooling Water System (RBCCW)

The heat loads on the RBCCW do not increase significantly by power uprate since 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 will not result in any significant increase in drywell cooling loads. The flow rates in the systems cooled by the RBCCW (e.g. recirculation and RWCU pumps cooling) will not change due to power uprate, therefore, are not affected by power uprate. The operation of the remaining equipment cooled by the RBCCW (e.g. instrument air compressors) is not power dependent and is not affected by power uprate. The RBCCW system contains enough redundancy in pumps and heat exchangers to assure that adequate heat removal capability is always available. Therefore, sufficient heat removal capacity is available to accommodate the increase in heat load due to power update.

6.4.4 Turbine Building Closed Cooling Water System (TBCCW)

The heat loads on the TBCCW which are power dependent and are increased by power uprate are those related to the operation of the turbine-generator. The remaining TBCCW heat loads are not strongly dependent upon reactor power and will not increase significantly. The TBCCW, like the RBCCW, contains enough redundancy to assure that adequate heat removal capability is always available. Therefore, sufficient cooling capacity for uprated operation is available.

6.4.5 Ultimate Heat Sink

The ultimate heat sink (UHS) for Brunswick Units 1 and 2 is the Cape Fear River. The cooling water intake for the Brunswick units is located on the river approximately five miles from where the river enters the Atlantic Ocean. The Brunswick units discharge into the Atlantic Ocean. Because of the proximity of the cooling water intake to the Atlantic Ocean, a dry season will not have an impact on the volume of water available in the UHS.

Power uprate does not impact the temperature of the water drawn from the river. Discharge temperatures to the Atlantic Ocean may increase in proportion to the 5% increase in decay heat; however, the impact on the Atlantic Ocean due to the increase in temperature is negligible.

None of the present limits for plant environmental releases such as UHS temperature or plant vent radiological limits will be exceeded as a result of power uprate.

6.5 Standby Liquid Control System (SLCS)

6.6 Power Dependent HVAC

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

The areas which will be affected by power uprate are in the drywell, reactor building, and turbine building. Other areas are unaffected by power uprate since process temperatures remain relatively constant. The heat rejection loads in the drywell are expected to increase less than 1% due to power uprate, which is conservatively estimated to increase the drwyell air temperature by less than 3°F. The room coolers serving the Core Spray, RHR, HPCI and RCIC pump rooms in the Reactor Building have adequate cooling margin to handle the minor heat load increases expected due to power uprate. In the remaining areas of the reactor building, temperature increases will not be significant. In the turbine building, the heat rejection loads are conservatively estimated to increase temperature by less than 1°F.

The heat load increases discussed above are minor and within the design capability of the existing systems; therefore, the design of the HVAC is not adversely affected by power uprate.

6.7 Fire Protection

Operation of the plant at the uprate power level does not affect the fire suppression or detection systems. There are no physical plant configuration or combustible load changes resulting from the uprate. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and are adequate for the uprated conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire protection systems and analyses are not affected by plant uprate.

6.8 Other Systems Reviewed for Impact By Power Uprate

In addition to the systems listed in Table J-1 of Reference 1 and not addressed elsewhere, the following systems have been evaluated and are not affected by operation of the plant at the uprated power level:

- Rod Control
- Reactor Building Sampling
- Post Accident Sampling
- Torus Drain
- Auxiliary Boiler
- Condensate Makeup
- Turbine Building Sampling
- Screen Wash
- Annunciator
- Turbine-Generator Lube Oil
- Gland Seal and Steam Seal
- Exhaust Hood Spray
- Turning Gear
- Hydrogen Seal Oil
- Diesel Generator
- Diesel Generator Fuel Oil
- Diesel Generator Lube Oil
- Diesel Generator Jacket Water and Demineralizer Water
- Diesel Generator Intake/Exhaust
- Diesel Generator Starting Air
- Emergency AC Lighting
- Emergency DC Lighting
- Auxiliary Control Board
- Instrument Air
- Service Air
- Pneumatic Nitrogen
- Hydrogen Supply
- Carbon-Dioxide Supply
- Fire Detection
- Lube Oil Storage and Transfer
- Fuel Oil
- Caustic
- Acid
- Radwaste Sampling
- Refueling
- Penetration Cooling

6.9 Systems With Minimal Impact

In addition to the systems listed in Reference 1, some systems are affected in a very minor way by operation of the plant at the uprated power level. These systems were evaluated and the effects are insignificant to the design or operation of the system and equipment. Changes to these systems to accommodate power uprate will be made as needed.

- Nuclear Steam Supply Shutoff
- Extraction Steam
- Generator Gas
- Stator Cooling Water
- ERFIS Computer
- Process Computer
- Main Control Board
- Hydrogen Water Chemistry

6.10 References

1. GE Nuclear Energy, *Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDO-31897, Class I (Non-proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.

Table 6-1
UPRATED PLANT ELECTRICAL DESIGN CHARACTERISTICS

Gross Generator Output (1.5/3.5 in. Hg vacuum)	895/867 MWe
Plant Net Output (1.5/3.5 in. Hg vacuum)	875/848 MWe
Rated Generator Voltage	24 kVAC
Apparent Power Output (Maximum, 0.95 Power Factor)	942 MVA
Corresponding Current Output	22.6 kA
Isolated Phase Bus Duct. Rating:	25 kA
Delta Leg to Main Transformers	14.45 kA
Unit Auxiliary Transformer Self-Cooled	1.5 kA
Main Transformer Rating	960 MVA

Table 6-2
FUEL POOL COOLING

	Current Power	Uprated Power	UFSAR Allowable Value (°F)
Case 1: Core Unload			
Maximum Heat Load (MBtu/hr)	29.2	31.3	
Maximum Calculated Pool Temperature (°F)	124.6	130.6	150
Case 2: Last Refueling			
Maximum Heat Load (MBtu/hr)	14.1	15.1	
Maximum Calculated Pool Temperature (°F)	145.2	148.8	150

Table 6-3
EFFLUENT DISCHARGE COMPARISON

	State Limit	Current*	Power Uprate**
Cooling Water Flow/Unit			
Winter			
Cubic feet per second	922	922	922
Gallon per minute	413,820	413,820	413,820
Summer			
Cubic feet per second	1105	1105	1105
Gallon per minute	495,960	495,960	495,960
Condenser Temperature Rise			
Winter			
Degree F	46.0	27.7	29.1
Degree C	25.6	15.4	16.2
Summer			
Degree F	30.0	23.2	24.4
Degree C	16.7	12.9	13.6

* Most conservative value for each observed parameter, and do not represent currently observed conditions.

** Calculated

7.0 POWER CONVERSION SYSTEMS

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 10, that applies to power uprate.

The power conversion systems were designed to utilize the energy available from the nuclear steam supply system and to accept the system and equipment flows resulting from continuous operation at 106% of rated steam flow.

7.1 Turbine-Generator

Power uprate increases the steam flow by 5.8%. The generator engineering evaluation shows the turbine-generator is capable of achieving the proposed capability without any modifications to the turbine-generator and its associated equipment.

Both the turbine and generator were originally sized to allow for higher flows. This provides control of important variables, such as steam inlet pressure, and the turbine-generator, as presently designed, can achieve the uprated conditions.

Under power uprate, each Brunswick units will utilize a monoblock rotor design. The dominant mode of wheel failure and missile generation was brittle fracture; however, the monoblock rotor design eliminates the brittle fracture mode of failure, so the probability of burst and missile is negligible and not a safety concern.

The load factors and stresses on all stages - high-pressure and low-pressure - of the turbine were reviewed. It was found that all stages will remain within design limits and are acceptable for operation at the uprated steam conditions.

A mechanical review of the turbine rotors was performed to evaluate steady-state, vibration, and upset stress conditions that are affected by the uprated steam conditions and loadings. The review concluded that power uprate operation is within design limits.

7.2 Condenser Air Removal System and Steam Jet Air Ejectors

The design of the condenser air removal system is not adversely affected by power uprate and no modification to the system is required. The physical size and design of the primary condenser and evacuation time are the main factors in establishing the capabilities of the mechanical vacuum pumps. These parameters do not change. Since flow rates will not change, there will be no change to the two-minute holdup time in the pump discharge line routed to the reactor building vent stack.

The flow to the steam jet air ejectors (SJAE) increases approximately 2.8% because of the uprate; however, the design capacity of the SJAE will not be affected by the uprate since they originally were designed for operation at significantly greater than warranted flows.

7.3 Turbine Steam Bypass

The turbine bypass capacity of Unit 1 is 25% of the warranted reactor steam flow and the bypass capacity of Unit 2 is 85% of the warranted reactor steam flow. The small pressure increase due to power uprate is within the original design capability of the turbine bypass system and increases the mass flow.

7.4 Feedwater and Condensate Systems

The feedwater and condensate systems are designed to provide a reliable supply of feedwater at the temperature, pressure, quality, and flow rate as required by the reactor. These systems are not safety-related; however, their performance has a major effect on plant availability and capability to operate at the uprated condition. For power uprate, the feedwater and condensate systems meet the following performance criteria:

- The systems provide a reliable supply of feedwater at the uprated dome pressure with sufficient capacity to supply the steady-state feedwater flow demanded at the uprated condition.
- The systems have the capacity to provide at least 110% of the uprated feedwater flow. This assures that the plant remains available during water level transients, avoids scrams, and minimizes challenges to plant safety systems.
- The feedwater system is capable of providing adequate uprated feedwater flow at the pressure selected for uprated operation with one feedwater pump tripped. This operational performance criterion assures that the plant remains available during a feedwater pump trip and subsequent recirculation system runback.
- The runout capacity of the feedwater system in the limiting pump alignment does not exceed the performance capacity assumed in the transient analyses.

These performance criteria were based on an assessment of the capability of the condensate and feedwater system equipment to remain within the design limitations of the parameters listed below:

- Pump NPSH
- Ability to avoid suction pressure trip
- Flow capacity
- Bearing cooling capability
- Rated motor horsepower
- Full load motor amps
- Vibration

7.4.1 Normal Operation

The condensate and feedwater systems were originally designed for steady-state operation at 105% (valves wide open) of warranted steam flows. UFSAR Figure 10.2.2-1 shows the heat balance at 100% warranted steam flow. Operation at the uprated power level does not significantly affect operating conditions of these systems. Discharge pressure at the condensate and condensate booster pumps decreases due to the pump head characteristics at increased flows. Discharge pressure at the feedwater pumps increases to compensate for the increase in reactor pressure and feedwater friction losses due to higher flows. The feedwater pump steam turbine automatically increases its speed, as necessary, to accomplish this. The condensate and feedwater systems were originally designed to meet a transient flow requirement of 115% of the original warranted steam flow during reactor flow transients. During steady-state conditions, the condensate and feedwater systems have sufficient NPSH for all of the pumps to operate without cavitation in the uprated conditions. Sufficient margins exist between the calculated minimum pump suction pressures experienced and minimum pump suction pressure based on NPSH to provide adequate trip margins during steady-state conditions. The existing feedwater design pressure and temperature requirement are adequate.

The feedwater heaters and associated regulating valves were originally designed for warranted flow conditions and, therefore, are adequate for the uprated conditions.

7.4.2 Transient Operation

Following a single feedwater pump trip, the reactor recirculation system will runback recirculation flow such that the steam production rate is within the flow capacity of the remaining feedwater pump. The runback setting is set low enough to prevent a reactor low water level scram, but high enough to avoid the potential power/flow instability regions.

7.4.3 Condensate Filter Demineralizers

The impact of power uprate on the condensate filter demineralizers (CFDs) was reviewed. In summary, the system is adequate for uprate operation, but will experience a reduction in CFD run times inversely proportional to the increase in flow; however, the reduced run times will be acceptable (refer to Section 8 for impact on radwaste systems).

7.4.4 Condensate Deep Bed Demineralizers

The impact of power uprate on the condensate deep bed demineralizers was reviewed. In summary, the system is adequate for uprate operation. The system will see an increase in condensate flow due to uprate, but has sufficient capacity to accommodate the flow.

8.0 RADWASTE SYSTEMS AND RADIATION SOURCES

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapters 11 and 12, that applies to power uprate.

8.1 Liquid Waste Management

The liquid radwaste system collects monitors, processes, stores, and returns processed radioactive waste to the plant for reuse or for discharge.

The single largest source of liquid waste is from the backwash of the condensate demineralizers. With power uprate, the average time between backwash/precoat will be reduced slightly. This reduction does not affect plant safety.

The floor drain collector subsystem and the waste collector subsystem both receive periodic inputs from a variety of sources. Neither subsystem is expected to experience a significant increase in the total volume of liquid waste due to operation at the uprated condition.

The activated corrosion products in liquid wastes are expected to increase approximately 5% (proportional to the power uprate); however, the total volume of processed waste is not expected to increase appreciably since the only significant increase in processed waste is due to the more frequent backwashes of the condensate demineralizers. Based on a review of plant operating effluent reports and the slight increase expected from power uprate, it is concluded that the requirements of 10CFR20 and 10CFR50, Appendix I will be met. Therefore, power uprate will 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 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, SCTS (Section 4.4), and various building ventilation systems. The systems are designed to meet the requirements of 10CFR20 and 10CFR50, Appendix I.

Noncondensible radioactive gas from the main condenser, along with air in-leakage, normally contains activation gases (principally N-16, O-19 and N-13) and fission product radioactive noble gases. These are the major source of radioactive gas (greater than all other sources combined). The noncondensible gas is continuously removed from the main condensers by the steam jet air ejectors (SJAE) which discharge into the offgas system.

Building ventilation systems control airborne radioactive releases by using combinations of devices such as HEPA and charcoal filters, and radiation monitors that signal automatic isolation dampers or trip supply and/or exhaust fans, or by maintaining negative air pressure, where required, to limit migration of gases. The activity of airborne effluents released through building vents is not expected to increase significantly with power uprate. This is because the amount of fission products released into the coolant depends on the number and nature of the fuel rod defects, which are approximately linear with respect to core thermal power; however, the release limit from the core does not change as a function of rated power level. There are no significant environmental effects.

8.2.1 Offgas System

Core radiolysis (i.e., formation of H_2 and O_2) will increase linearly with core power. The original design flow was about 40% above the actual observed flow of H_2. The operational increases in H_2 and O_2 due to power uprate remain within the design flow capacity of the system.

The system radiological release rate is not changed with rated power level. Therefore, power uprate will not affect the offgas system design or operation.

8.3 Radiation Sources in the Reactor Core

8.3.1 Operation

8.3.2 Post-Operation

8.4 Radiation Sources in the Coolant

8.4.1 Coolant Activation Products

8.4.2 Activated Corrosion Products

8.4.3 Fission Products

8.5 Radiation Levels

8.5.1 Normal Operation

As a result of power uprate, normal operation radiation levels increase approximately 5%. 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, since it is offset by conservatism in the original design, source terms used, and analytical techniques.

8.5.2 Post-Operation

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 near the reactor water piping and liquid radwaste equipment, the increase could be slightly higher. 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, since it is offset by conservatism in the original design, source terms used, and analytical techniques. Regardless, individual worker exposures will be maintained within acceptable limits by the site Health Physics program, which controls access to radiation areas.

8.5.3 Post-Accident

The change in core inventory resulting from power uprate (Section 8.3) is expected to increase post-accident radiation levels by no more than the percentage increase in power level. The slight increase in the post-accident radiation levels has no significant effect on the plant, or on the habitability of the Technical Support Center (TSC) or Emergency Operations Facility (EOF).

8.5.4 Offsite Doses (Normal Operation)

The normal offsite doses at the uprated power level remain below the limits of 10CFR20 and 10CFR50, Appendix I.

CP&L's review of areas requiring post-accident occupancy (per NUREG-0737 Item II.B) concluded that access to the reactor building is not needed for accident mitigation, and hence this is not a concern for power uprate operation.

9.0 REACTOR SAFETY PERFORMANCE EVALUATIONS

This section primarily focuses on the information requested in Regulatory Guide 1.70, Chapter 15, that applies to power uprate.

9.1 Reactor Transients

The USAR evaluates the effects of a wide range of potential plant transients. Disturbances of the plant caused by a malfunction or a single failure of equipment or the operator are investigated according to the type of initiating event. The generic guidelines for BWR power uprate (Appendix E of Reference 1) identify the limiting event(s) to be considered in each category of events. The generic guideline also identified the analytical methods, the operating conditions that are to be assumed, and the criteria that are to be applied.

The following sections address each event and provide a summary of the resulting transient safety evaluations. The results given here are for a representative (Brunswick Unit 1 Cycle 8) core, and show the overall capability of the design of both units to meet all transient safety criteria for uprated operation.

Table E-1 of Reference 1 provides the specific events to be analyzed for power uprate, the power level to be assumed, and the computer models to be used. The power uprate analysis used the GEMINI transient analysis methods listed there. The Reference 1 analyses encompass the limiting events for Brunswick Units 1 and 2.

The reactor operating conditions that apply most directly to the transient analysis are summarized in Table 9-1. They are compared to the conditions used for the UFSAR and the fuel cycle analyses. The Unit 1 Cycle 8 core was used as the representative fuel cycle for power uprate. Unit 1 was selected for the transient evaluations because it bounds Unit 2. Most of the transient events are analyzed at the full uprated power and the 105% core flow operating point, which bounds the power/flow map shown in Figure 2-1. Direct or statistical allowance for 2% power uncertainty is included in the analysis. The Safety Limit MCPR (SLMCPR) in Table 9-1 was used to calculate the MCPR Operating Limits provided for the analyzed events. For all pertinent events, 2 SRVs are assumed to be out-of-service.

The effect of power uprate on the SLMCPR is generically evaluated in Section 3.4 of Reference 2.

9.2 Design Basis Accidents

Plant specific radiological analyses were performed at uprated conditions for selected postulated accidents. The dose analyses were based on the plant parameters used in the UFSAR analyses and the assumptions provided in the applicable NRC Regulatory Guides. Dose calculations were performed at the current power level and at the uprated power level to determine the magnitude of the change attributed solely to the change in power. The analyses were performed using GE's standard models that were utilized on other power uprate projects.

The events reanalyzed were the Loss-of-Coolant-Accident (LOCA), the Main Steam Line Break Accident (MSLBA) outside containment, the Fuel Handling Accident (FHA), and the Control Rod Drop Accident (CRDA). The whole body and thyroid dose were calculated at the exclusion area boundary, low population zone (LPZ), and for the LOCA, in the main control room. Inputs to the analyses are presented in Tables 9-3 through 9-7. The doses resulting from the accidents analyzed are compared with the applicable dose limits in Tables 9-8 through 9-11 for the current power level and the uprated power level. The plant specific results for power uprate remain well below established regulatory limits.

Other accidents (non-LOCA) analyzed in the FSAR have similarly been generically reviewed in Section 5.2 of Reference 2, and remain below their regulatory limits for power uprate operation.

9.3 Special Events

9.3.1 Anticipated Transients Without Scram (ATWS)

9.3.2 Station Blackout (SBO)

Plant response and coping capabilities for a station blackout (SBO) event are impacted slightly by operation at the uprated power level due to the increase in the operating temperature of the primary coolant system, increase in the decay heat, and increase in the main steam safety relief valve setpoints. There are no changes to the systems and equipment used to respond to an SBO, nor is the required coping time changed.

The Brunswick response to a postulated SBO is to utilize the HPCI System with DC augmentation from the other unit to cope for four hours. Emergency diesel-generator and station battery performance are adequate for response to an SBO with power uprate operation.

The following areas contain equipment necessary to mitigate the station blackout event:

- Control, Diesel-Generator Basement and 480V EBUS Switchgear Rooms
- HPCI and RCIC Equipment Room
- ECCS Pipe Tunnel
- Containment

The temperature increases in the Control, Diesel-Generator Basement and 480V EBUS Switchgear Rooms are not affected by power uprate. The HPCI and RCIC equipment room temperatures and the ECCS pipe tunnel temperature will increase; however, significant margin exists to operability limits so that the operability of equipment in these locations is not affected.

Suppression pool temperature increases about 4°F and containment pressure about 2 psi, but the increases are small enough to not affect equipment operability.

Also, the condensate water requirement increases less than 3%; however, the current condensate storage tank design ensures that adequate water volume is available.

9.3.3 Appendix R

Brunswick calculations associated with 10CFR50 Appendix R have been reviewed for the 105% power uprate condition. The conclusions of the review are that operation of the plants at the 105% power level will not affect the ability of the safe shutdown systems to perform their intended function, and the minimum systems and equipment required for safe shutdown at power uprate do not change.

9.4 References

1. GE Nuclear Energy, *Generic Guidelines For General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDO-31897, Class I (Non-proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.

2. GE Nuclear Energy, *Generic Evaluations of General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-proprietary), March 1992; and Supplements.

Table 9-1
PARAMETERS USED FOR TRANSIENT ANALYSIS

Parameter	Units	UFSAR	Unit 1 Cycle 8 Analysis	Power Uprate
Rated Thermal Power	(MWt)	2436	2436	2558
Analysis Power*	(% Rated)	100	100	100
Analysis Dome Pressure	(psia)	1020	1020	1045
Analysis Turbine Pressure	(psia)	965	965	1000
Rated Steam Flow	(Mlb/hr)	10.47	10.47	11.08
Analysis Steam Flow	(% Rated)	100	100	100
Rated Core Flow	(Mlb/hr)	77	77	77
Rated Power Core Flow Range	(% Rated)	100	75-105	80-105
Analysis Core Flow+	(% Rated)	100	105	105
FW Temperature for Analysis	(°F)	420	420	425
Number of SRVs++		9	9	9
Analysis SRV Setpoints	(psig)	See Table 5-1	See Table 5-1	See Table 5-1
Assumed MCPR Safety Limit		1.07	1.07	1.07

* ODYN/GEMINI analysis at 100% of uprated power. All other analyses at 102% of uprated power.
\+ All analysis at maximum core flow unless explicitly noted otherwise.
++ The lowest pressure setpoint SRVs are assumed to be out of service for transient analysis.

Table 9-2
TRANSIENT ANALYSIS RESULTS FOR POWER UPRATE[3]

Event	Neutron Flux (% Rated)	Heat Flux (% Rated)	MCPR Delta CPR	MCPR Opt. A	MCPR Opt. B
Turbine Trip w/Bypass Failure	(1)	(1)	(1)	(1)	(1)
Generator Load Rejection w/Bypass Failure	804 [663]	129 [128]	0.27 [0.27]	1.39 [1.39]	1.35 [1.35]
Feedwater Controller Failure – Max Demand	580 [506]	127 [125]	0.22 [0.22]	1.34 [1.33]	1.31 [1.30]
Loss of Feedwater Flow	(1)	(1)	(1)	(1)	(1)
Inadvertent HPCI Activation	(1)	(1)	(1)	(1)	(1)
Loss of Feedwater Heating	–	–	0.12 [0.11]	1.19 [1.18]	1.19 [1.18]
Rod Withdrawal Error	–	–	0.09* [0.09]	1.16* [1.16]	1.16* [1.16]
Slow Recirculation Increase[2]	–	–	$MCPR_f$	$MCPR_f$	$MCPR_f$

* With Rod Block Monitor setpoint of 108%.

(1) Bounded by Generator Load Rejection w/Bypass Failure.

(2) The analysis of anticipated operational occurrences for BWR core reloads is generally performed assuming rated and/or maximum allowable core flow conditions. The purpose of the $MCPR_f$ curve is to adjust the MCPR operating limits when the plant is operating at less than rated core flow. This is necessary because some transient events are more severe (i.e., larger ΔMCPR) when initiated at reduced core flow. The $MCPR_f$ curve assures that the Safety Limit MCPR will not be violated should the most limiting (e.g., flow increase) transient occur.

(3) Values in brackets are the pre-uprate results for Unit 1, Cycle 8.

Table 9-3

ASSUMPTIONS FOR LOSS-OF-COOLANT ACCIDENT

Power (MWt)	2558
Power Multiplier Factor	1.02
Initial Inventory Fractions in Containment Atmosphere (%)	
Noble gases	100
Iodines	25
Primary Containment Leak Rate (%/day)	0.5
MSIV Leakage Rate (%/day)	0.0
Fraction of Containment Leakage Which Bypasses SGTS (%)	0.0
Holdup in Secondary Containment	None
Duration of Exfiltration During Secondary Containment Drawdown (min.)	0.0
SGTS Iodine Filter Efficiency (%)	99
ECCS Leakage in Secondary Containment	None
Release Height (m)	100
Site Boundary Distance (m)	914, 3219
X/Q at EA Boundary (sec/m^3)	
0-2 hours	2.0E-5
X/Q at LPZ (sec/m^3)	
0-8 hours	8.8E-6
8-24 hours	3.8E-6
24-96 hours	1.1E-6*
96-720 hours	3.5E-7
Thyroid Inhalation DCF (rem/Ci)	
I-131	1.49E+6
I-132	1.43E+4
I-133	2.69E+5
I-134	3.73E+3
I-135	5.60E+4

* RG 1.3 value is more conservative than the UFSAR value.

Table 9-3 (continued)

ASSUMPTIONS FOR LOSS-OF-COOLANT ACCIDENT

Fission Product Inventory (Ci/MW)

I-131	2.63E+4
I-132	3.85E+4
I-133	5.50E+4
I-134	6.06E+4
I-135	5.20E+4
Kr-83m	3.14E+3
Kr-85m	6.73E+3
Kr-85	3.02E+2
Kr-87	1.29E+4
Kr-88	1.83E+4
Kr-89	2.28E+4
Xe-131m	1.58E+2
Xe-133m	2.31E+3
Xe-133	5.53E+4
Xe-135m	1.04E+4
Xe-135	7.15E+3
Xe-137	4.85E+4
Xe-138	4.61E+4

Table 9-4

ASSUMPTIONS FOR LOCA CONTROL ROOM DOSE

Control Room Volume Treated (ft^3)	298,650
Control Room Volume Contributing to Gamma Dose (ft^3)	298,650
Chemical Form of Iodine	R.G. 1.3 Assumptions
Filtered In-leakage Flow (ft^3/min)	1000
Unfiltered In-leakage Flow (ft^3/min)	3000
Air Filter Efficiency (%)	
Organic	90
Particulate	95
Elemental	95
Filtered Recirculation Flow (ft^3/min)	1000
Unfiltered Recirculation Flow (ft^3/min)	38000
Recirculation Filter Efficiency (%)	
Organic	90
Particulate	95
Elemental	95
Occupancy Factor	1.0*
X/Q for Stack Release (sec/m^3)	
0-0.5 hour	3.3E-4
0.5-8 hours	1.8E-6
8-24 hours	1.1E-6
24-96 hours	2.0E-7
96-720 hours	2.7E-8

* Conservative assumption

Table 9-5

ASSUMPTIONS FOR MAIN STEAM LINE BREAK ACCIDENT

Power (MWt)		2558
Power Multiplier Factor		1.02
Initial Steam Dome Volume (ft^3)		4428
Iodine Concentration in Coolant (μCi/gm)		
Case 1:	I-131	0.073
	I-132	0.71
	I-133	0.50
	I-134	1.40
	I-135	0.73
	DE I-131	0.2
Case 2:	I-131	1.45
	I-132	14.1
	I-133	9.95
	I-134	27.9
	I-135	14.5
	DE I-131	4.0
MSIV Closure Time (sec)		10.5
Coolant Discharged from Break (lbm)		96,505
Fraction of Iodine in Released Coolant Assumed Airborne (%)		100
Noble Gas Release Rate prior to MSIV Closure (μCi/s at 30 min decay)		300,000
Holdup in Turbine Building		No
Release Height (m)		30
Meteorological Condition		Fumigation
X/Q at EA (sec/m^3)		8.4E-4
X/Q at LPZ (sec/m^3)		1.7E-4

Table 9-6

ASSUMPTIONS FOR CONTROL ROD DROP ACCIDENT

Power (MWt)	2558
Power Multiplier Factor	1.02
Radial Power Peaking Factor	1.5
Number of Failed Fuel Rods	850
Number of Fuel Rods per Bundle	62
Number of Bundles	560
Fuel Rod Plenum Activity Fractions (%)	
Noble gases	10
Iodines	10
Fraction of Fuel in Failed Rods Assumed to Melt (%)	0.77
Fraction of Inventory Released from Melted Fuel (%)	
Noble gases	100
Iodines	50
Fraction of Released Activity Transported to Condenser (%)	
Noble gases	100
Iodines	10
Decay Prior to Release	None
Fraction of Iodine Washed Out in Condenser (%)	90
Leak Rate from Condenser (%/day)	1.0
Release Period from Condenser (hrs)	24
Holdup in Turbine Building	No
Release Height (m)	0
X/Q at EA Boundary (sec/m^3)	
0-2 hours	8.4E-4
X/Q at LPZ (sec/m^3)	
0-8 hours	1.7E-4
8-24 hours	2.4E-5

Table 9-7

ASSUMPTIONS FOR FUEL HANDLING ACCIDENT

Power (MWt)	2558
Power Multiplier Factor	1.02
Decay Time After Shutdown (hrs)	24
Number of Failed Fuel Rods	104
Fuel Rod Plenum Activity Fractions (%)	
Noble gases (except Kr-85)	10
Kr-85	30
Iodines	10
Number of Fuel Rods per Bundle	62
Number of Bundles	560
Radial Power Peaking Factor	1.5
Decontamination Factor in Water	
Noble gases	1
Iodines	100
Time Period for Release of Activity from Reactor Building (hr)	<2
SGTS Iodine Filter Efficiency (%)	99
Release Height (m)	100
X/Q at EA Boundary (sec/m^3)	
0-2 hours	2.0E-5
X/Q at LPZ (sec/m^3)	
0-2 hours	8.8E-6

Table 9-8
LOCA RADIOLOGICAL CONSEQUENCES

Location	Current Power	Uprated Power	Limit
Exclusion Area:			
Whole Body Dose, rem	0.27	0.29	≤ 25
Thyroid Dose, rem	1.02	1.07	≤ 300
Low Population Zone:			
Whole Body Dose, rem	0.31	0.32	≤ 25
Thyroid Dose, rem	3.83	4.02	≤ 300
Control Room:			
Whole Body Dose, rem	0.12	0.13	≤ 5
Thyroid Dose, rem	3.11	3.26	≤ 30
Beta Dose, rem	0.83	0.88	≤ 30

Table 9-9

MSLBA RADIOLOGICAL CONSEQUENCES

Location/Quantity	Current Power	Uprated Power	Limit
Case 1:			
Iodine concentration in coolant = 0.2 μCi/gm dose-equivalent I-131.			
Exclusion Area:			
Whole Body Dose, rem	0.065	0.065	≤ 2.5
Thyroid Dose, rem	3.83	3.83	≤ 30
Low Population Zone:			
Whole Body Dose, rem	0.014	0.014	≤ 2.5
Thyroid Dose, rem	0.793	0.793	≤ 30
Case 2:			
Iodine concentration in coolant = 4.0 μCi/gm dose-equivalent I-131.			
Exclusion Area:			
Whole Body Dose, rem	1.25	1.25	≤ 25
Thyroid Dose, rem	76.6	76.6	≤ 300
Low Population Zone:			
Whole Body Dose, rem	0.259	0.259	≤ 25
Thyroid Dose, rem	15.9	15.9	≤ 300

Table 9-10
FHA RADIOLOGICAL CONSEQUENCES

Location/Quantity	Current Power	Uprated Power	Limit
Exclusion Area:			
Whole Body Dose, rem	0.033	0.034	≤ 6
Thyroid Dose, rem	0.034	0.036	≤ 75
Low Population Zone:			
Whole Body Dose, rem	0.014	0.015	≤ 6
Thyroid Dose, rem	0.015	0.016	≤ 75

Table 9-11

CRDA RADIOLOGICAL CONSEQUENCES

Location/Quantity	Current Power	Uprated Power	Limit
Exclusion Area:			
Whole Body Dose, rem	0.093	0.097	≤ 6
Thyroid Dose, rem	1.29	1.36	≤ 75
Low Population Zone:			
Whole Body Dose, rem	0.039	0.041	≤ 6
Thyroid Dose, rem	1.12	1.17	≤ 75

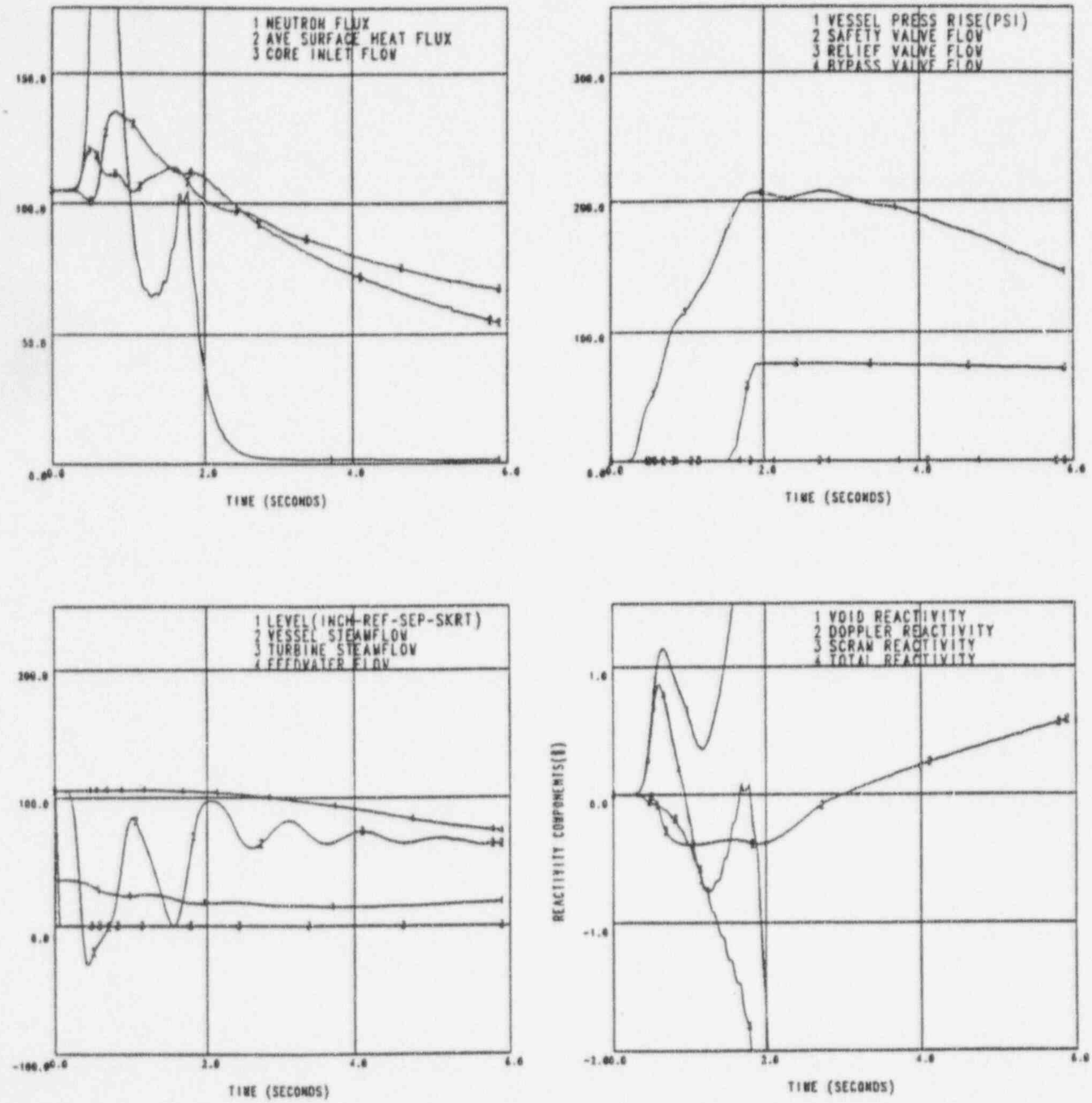

Figure 9-1. Generator Load Rejection with BP Failure @ 100% Uprated Power/105% Core Flow

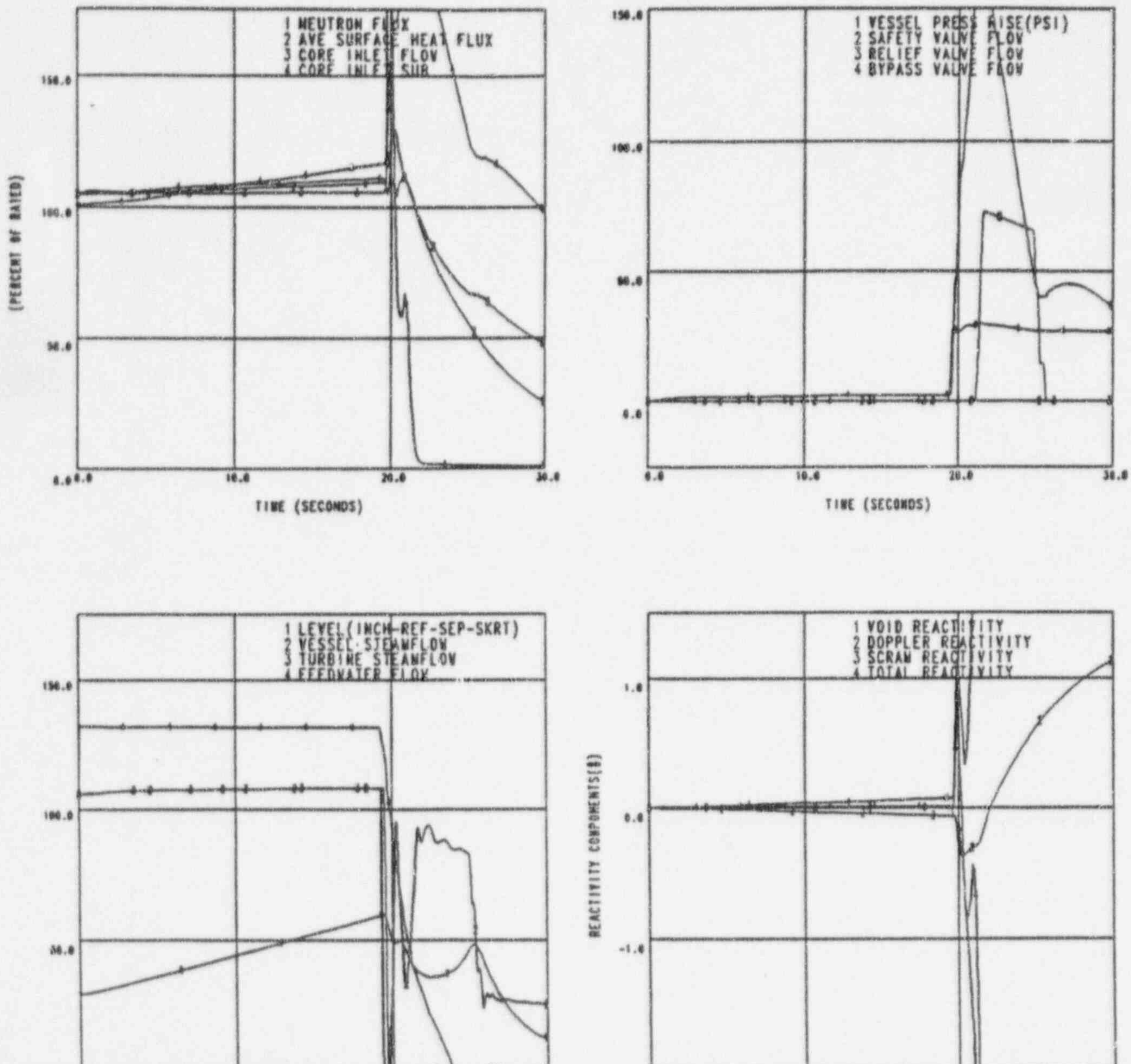

Figure 9-2. Feedwater Controller Failure - Maximum Demand @ 100% Uprated Power/105% Core Flow

10.0 ADDITIONAL ASPECTS OF POWER UPRATE

10.1 High Energy Line Break (HELB)

Operation at an uprated level requires a small increase in the RPV dome operating pressure to supply more steam to the turbine. The slight increase in the vessel pressure and temperature results in a small increase in the mass and energy release rates following high energy line breaks (HELBs). Evaluation of these piping systems determined that there is no change in the postulated break locations identified in the UFSAR.

10.1.1 Temperature, Pressure, and Humidity Profiles

At the uprated power level, HELBs outside the primary containment cause the subcompartment pressure and temperature profiles to increase. The relative humidity change is negligible. For Brunswick, the increase in the blowdown rate due to power uprate is small (3% or less).

The HELB analysis evaluation was made for all systems evaluated in the UFSAR. The evaluation shows that the affected building and cubicles that support the safety-related function are designed to withstand the resulting pressure and thermal loading following a HELB. The equipment and systems that support the safety-related function are also qualified for the environmental conditions imposed upon them, as described in Section 10.2.

10.1.1.1 Main Steam System Line Break

The critical parameter affecting the HELB analysis relative to the power uprate is the small increase in reactor vessel dome pressure. As a result of the power uprate, the blowdown rate increases by a small amount (approximately 1.5%), and the differential pressure between sub-compartments and peak temperature in the steam tunnel increases a small amount. Profiles for surrounding areas are not significantly affected and are acceptable.

10.1.1.2 High Pressure ECCS Line Break

The steamline break in the high pressure ECCS pump/turbine room is the limiting break for structural design and equipment qualification. The short-term peak pressure differential between rooms increases by an amount which is structurally acceptable. The increase in break flow rate at the uprated power level is small (approximately 0.1%).

10.1.1.3 Reactor Core Isolation Cooling System Line Break

The steamline break in the RCIC pump/turbine and torus rooms are the limiting breaks for structural design and equipment qualification. The original analysis was performed with conservative model assumptions. Inclusion of more realistic assumptions reduces the blowdown rate by more than enough to compensate for the uprate increase. Therefore, the previous HELB analysis is bounding for the uprated core condition.

10.1.1.4 Reactor Water Cleanup System Line Breaks

The RWCU 6-inch supply line break and the RWCU 4-inch return line break are the limiting breaks for structural design and equipment qualification. The original analysis for the 6-inch line break was performed with conservative model assumptions. For the 6-inch line break, these conservatisms more than offset the effect of increase in the pressure, so the original HELB analysis is bounding for the uprated core condition. For the 4-inch line break, the break flow increases slightly (approximately 2%) for the power uprate condition.

10.1.1.5 Control Rod Drive System Line Break

The CRD pipe rupture analysis is not affected by power uprate. The piping contains highly subcooled fluid, so there is no adverse impact.

10.1.2 Pipe Whip and Jet Impingement

This section addresses the evaluation of the effects of jet impingement from high energy lines addressed in UFSAR Section 3.6.

As shown in the previous sections, there are no new break locations as a result of power uprate. In addition, the maximum increase in break flow is 2% (for the RWCU 4-inch line break). Calculations supporting the dispositions of potential targets of pipe whip and jet impingement from the postulated HELBs have sufficient conservatism included to cover the slight increase in break flows. The existing pipe whip restraints and jet impingement shields, and their supporting structures, have been determined to be adequate for safe shutdown effects at the uprated conditions.

10.1.3 Moderate Energy Line Break (MELB)

Operation at an uprated level requires a small increase in the reactor vessel pressure during full power operation. The high pressure ECCS, RCIC, RWCU, and CRD Systems water piping are the only moderate energy lines affected. Of these, the CRD System is the only one which has the limiting crack in any plant area, even at the uprated conditions. The MELB evaluation for the CRD piping was originally done at the system design pressure, which is unaffected by the power uprate. Therefore, the MELB evaluation for the plant is not impacted.

10.2 Equipment Qualification (EQ)

10.2.1 EQ of Electrical Equipment

The safety-related electrical equipment is evaluated to assure qualification for the normal and accident conditions expected in the area where the devices are located. Conservatisms in accordance with DOR Guidelines/IEEE-323 (as applicable) were applied to the environmental parameters as required.

10.2.1.1 Inside Containment

Equipment qualification for safety-related electrical equipment located inside the containment is based on main steamline break and/or DBA LOCA conditions and the resultant temperature, pressure, humidity and radiation consequences. It includes the environments expected to exist during normal plant operation. The current accident and normal conditions for temperature, pressure, and humidity are unchanged for the uprated power conditions. Based on uprate evaluations for other BWRs, the current radiation levels under normal plant conditions are expected to increase 5%, and under accident conditions are expected to increase 8-10%.

The qualification of any equipment located within the containment which is affected by the higher accident radiation level will be resolved either by (1) refined radiation calculations (location specific), (2) slightly reducing qualified life, (3) assessing the conservatisms in the qualification basis, or (4) a reassessment of the impact of the radiation on the materials of construction.

10.2.1.2 Outside Containment

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from a main steamline break in the pipe tunnel, or other HELBs, whichever is limiting for each plant area. The accident temperature, pressure and humidity conditions resulting from a LOCA do not change with the power level; however, some of the HELB profiles do increase by a small amount (Section 10.1). Maximum accident radiation levels used for qualification of equipment outside containment are from a DBA/LOCA.

The accident temperature, pressure and radiation levels increase due to operation at the uprated power level, as shown in Table 10-1. The normal temperature, pressure, and humidity conditions do not change as a result of power uprate. The normal radiation levels were evaluated for other BWR uprates, and, based on those evaluations, are expected to increase slightly (Table 10-1).

The qualification of any equipment impacted by the uprated conditions will be resolved either by (1) refined radiation calculations (location specific), (2) slightly reduced qualified life, (3) assessing the conservatisms in the qualification basis, or (4) a reassessment of the impact of the radiation on the materials of construction.

10.2.2 EQ of Mechanical Equipment with Non-Metallic Components

Operation at the uprated power levels slightly increases the normal process temperature (<5°F). The accident radiation level and normal radiation level also increase slightly due to uprate, and were evaluated as discussed in Section 10.2.1. Reevaluation of the safety-related mechanical equipment with non-metallic components identified some equipment potentially impacted by the uprated radiation conditions. The qualification of this equipment is resolved either by (1) refined radiation calculations (location specific), (2) slightly reduced qualified life, (3) assessing the conservatisms in the qualification basis, or (4) a reassessment of the impact of the radiation on the materials of construction.

10.2.3 Mechanical Component Design Qualification

The mechanical design of equipment/components (pumps, heat exchangers, etc.) in certain systems is affected by operation at the uprated power level due to slightly increased temperatures (<5°F), pressure (<3 psi), and in some cases, flow (≤5%). The revised operating conditions do not significantly affect the cumulative usage fatigue factors of mechanical components.

SRV loads are addressed in Sections 3.5 and 4.1.2.2.

10.3 Required Testing

Compared to the initial startup program, and consistent with the NRC approved generic power uprate guidelines in Reference 1, power uprate requires limited startup tests. As applicable to this plant's design, testing for power uprate will be consistent with the descriptions in Section 5.11.9 and Appendix L, Section L.2 of Reference 1. Specifically, the following testing will be performed at the time of implementation of power uprate:

- Surveillance testing will be performed on the instrumentation that requires recalibration for power uprate.

- Steady-state data will be taken at points from 90% up to the previous rated thermal power, so that operating performance parameters can be projected for uprated power before the previous power rating is exceeded.

- Power increases beyond the previous rating will be made along an established flow control/rod line in increments of ≤ 3%. Steady-state operating data will be taken and evaluated at each step. Indicated core power will be rescaled in terms of the new rating before exceeding the current rating.

- Control system checks will be performed for the feedwater reactor water level controls and pressure controls. These operational checks will be made at the previous rated power condition and at each power increment, to show acceptable adjustments and operational capability. The same performance criteria shall be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program.

- Fuel thermal margin will be demonstrated prior to and during power ascension to the uprated power level.

10.4 Shutdown and Refueling Requirements

There are no additional shutdown and refueling requirements as a result of power uprate. The shutdown and refueling systems that are impacted by power uprate (Shutdown Cooling, Fuel Pool Cooling) have been addressed in previous sections of this report.

10.5 Operator Training

Additional training required to operate the plant in an uprated condition is expected to be minimal. 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.

10.6 Plant Life

The longevity of most equipment is not affected by power uprate. There are various plant programs (Equipment Qualification, Flow Accelerated Corrosion) that deal with age-related components. These programs will not change as a result of power uprate. In addition, the Maintenance Rule will provide the oversight for the remaining mechanical and electrical components, important to plant safety, to guard against age-related degradation. The plant Nuclear Steam Supply System (the reactor pressure vessel, reactor internals, and primary coolant pressure boundary) are evaluated in Sections 3.3.2, 3.3.3 and 3.5.

10.7 Individual Plant Evaluation (IPE)

The Brunswick units use a probabilistic risk assessment (PRA) to comply with the IPE requirement. The PRAs of typical BWRs are evaluated in Supplement 2 of Reference 2, to generically determine the effects of a 5% power uprate on the results of BWR PRAs. This evaluation shows that a 5% power uprate for BWRs does not significantly affect plant safety. The evaluation results are based on an assessment of the effects by power uprate on the key PRA parameters that have the potential to affect the core damage frequency. The changes to these parameters were found to be relatively insensitive to the small changes in reactor power, pressure and flow. Based on the above review of several published BWR PRAs, it is concluded that the calculated core damage frequencies for BWRs will not significantly change due to power uprate.

The Brunswick PRA was reviewed against the bases and conclusions of the above generic evaluation. This review concludes that the conclusions of the generic evaluation in Reference 2 are applicable to Brunswick.

10.8 References

1. GE Nuclear Energy, *Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDO-31897, Class I (Non-proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.

2. GE Nuclear Energy, *Generic Evaluations of General Electric Boiling Water Reactor Power Uprate,* Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-proprietary), March 1992; and Supplements.

Table 10-1

ADDITIONAL ASPECTS OF POWER UPRATE

Normal uprated plant operation radiation increase assumption inside containment for EQ	5%
Accident uprated radiation increase assumption inside containment for EQ	8%
Accident uprated temperature increase assumption outside containment for EQ	1-3°F
Accident uprated pressure increase assumption outside containment for EQ	1-2 psi
Accident uprated radiation increase assumption outside containment for EQ	5%
Normal uprated radiation level increase assumption outside containment for EQ	3%
Uprate normal process temperature increase assumption for EQ of mechanical equipment with non-metallic components	<5°F
Uprate normal process radiation level increase assumption for EQ of mechanical equipment with non-metallic components	5%
Uprate accident radiation level increase assumption for EQ of mechanical equipment with non-metallic components	8%

11.0 LICENSING EVALUATIONS

11.1 Evaluation of Other Applicable Licensing Requirements

The analysis, design, and implementation of power uprate was reviewed for compliance with the plant's original licensing basis and for compliance with new regulatory requirements and operating experience in the nuclear industry. A generic review of the BWR power uprate program for compliance with regulatory requirements is provided in Reference 1. Plant-unique evaluations have been performed for the subjects addressed below. These subjects are identified as requiring plant unique confirmation in Reference 1.

11.1.1 NRC and Industry Communications

The issues from the following subjects are either generically evaluated in Reference 1, or are evaluated on a plant-specific basis as part of the power uprate program. These evaluations conclude that every issue is either (1) not affected by power uprate, (2) already incorporated into the power uprate program, or (3) bounded by the power uprate analyses.

- Code of Federal Regulations (CFRs)
- Regulatory Guides
- Generic Letters
- TMI Action Items
- Action Items (Formerly Unresolved Safety Issues)
- New Generic Issues
- Information and Enforcement Circulars (IEC)
- Inspection and Enforcement Notices (IENs)
- Inspection and Enforcement Bulletins (IEBs)
- GE Services Information Letters (SILs)
- GE Rapid Information Communication Service Information Letters (RICSILs)

11.1.2 Plant-Unique Items

Plant-unique items whose previous evaluations could be affected by operation at the uprated power level have been reviewed. These are the NRC and Industry communications listed above, Safety Evaluations (SEs) for work in progress and not yet integrated into the plant design basis, and Temporary Modifications (TM) which are not permanent changes, but could have been reviewed prior to the power uprate and still exist after uprate implementation. These items have been reviewed for possible impact by the power uprate, and were found to be either acceptable for uprate, or were revised to reflect the uprated conditions.

11.1.2.1 Safety Evaluations

Safety evaluations (SEs) for work in progress and SEs completed but not yet included in the UFSAR were reviewed for required changes due to uprated conditions.

11.1.2.2 Temporary Modifications

Temporary modifications that would be in effect after power uprate will be reviewed and revised, if necessary, to include uprated conditions.

11.1.2.3 Emergency Operating Procedures (EOP)

The plant EOPs will be reviewed for any effects of power uprate, and the EOPs will be updated, as necessary. This review will be based on Section 2.3 of Reference 1.

The EOPs for the Brunswick units are symptom based. Changes to the EOPs and the abnormal operating procedures (accident mitigation procedures) required for power uprate implementation are revisions to previously defined numerical values only (e.g., RPV high pressure scram setpoint value). The definition of these parameters has not been altered, only the numerical value of the parameter has changed. As such, the type, scope, and nature of the operator actions required for accident mitigation are unchanged. No new types of operator actions are necessary.

The response time for some operator actions during dynamic accident events at power uprate conditions may be slightly shorter when compared to the same events at pre-uprate conditions. However, the change in response time is not significant. The operating crew will still be able to successfully implement EOP actions. The type and scope of the operator actions remain unchanged. The accident mitigation strategy of the EOPs will not change. A procedure revision, other than numerical value changes, is not required.

11.2 Impact on Technical Specifications

Implementation of power uprate will require revision of a number of the Technical Specifications (TS). Table 11-1 contains a list of TS locations which should be changed to implement power uprate. A brief description of the nature of each change is also provided. The evaluations summarized in this report provide the justifications for these TS changes.

11.3 Environmental Assessment

A detailed environmental impact assessment is contained in the CP&L submittal.

The proposed power uprate does not require a change to the Environmental Protection Plan or constitute an unreviewed environmental question, since it does not involve either:

- A significant increase in any adverse environmental impact previously evaluated in the final statement, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board.

- A significant change in effluents or power level.

- A matter not previously reviewed and evaluated in the documents specified in item one above which may have a significant adverse environmental impact.

The evaluations also establish that power uprate qualifies for a categorical exclusion not requiring an environmental review in accordance with 10CFR51.22(c)(9) because it does not:

- Involve a significant hazard.

- Result in a significant increase in the amounts of any effluents that may be released offsite.

- Result in a significant increase in individual or cumulative occupational radiation exposure.

11.4 References

1. GE Nuclear Energy, *Generic Evaluations of General Electric Boiling Water Reactor Power Uprate*, Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-proprietary), March 1992; and Supplements.

Table 11-1

TECHNICAL SPECIFICATIONS AFFECTED BY POWER UPRATE

Location	Effect
Page 1-6	Revise definition of "rated thermal power".
Table 2.2.1-1	Revise APRM flow-biased simulated thermal power setpoint.
Figure 2.2.1-1	Rescale the figure for power uprate.
Table 3.3.2-2	Revise setpoints condenser vacuum, HPCI steam supply pressure, HPCI turbine exhaust diaphragm pressure, RCIC steam supply pressure, RCIC turbine exhaust diaphragm pressure, and reactor steam dome pressure.
Table 3.3.3-2	Revise setpoints for reactor steam dome pressure and RHR (LPCI) pump discharge pressure.
Table 3.3.4-2	Revise APRM flow-biased setpoint.
Table 3.3.6.1-2	Revise reactor vessel pressure setpoint.
Page 3/4, 4-16	Rescale figure for power uprate.
Page 3/4, 4-4	Revise SRV setpoints and tolerance.
Page 3/4, 4-21	Revise limit on reactor steam dome pressure.
Pages 3/4 6-1 through 6-5 and B3/4, 6-1 through 6-3	Revise pages and Bases to reflect containment evaluation power uprate conditions.
Page B 3/4, 11-6	Revise Bases (Section 3/4.11.2.7) to reflect new power level.

7/

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ENCLOSURE 3
BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324
LICENSE NOS. DPR-71 AND DPR-62

BRUNSWICK NUCLEAR PLANT
UNITS 1 & 2
POWER UPRATE OPERATION

NON-RADIOLOGICAL ENVIRONMENTAL ASSESSMENT

BRUNSWICK NUCLEAR PLANT
UNITS 1 & 2
POWER UPRATE OPERATION

NON-RADIOLOGICAL ENVIRONMENTAL ASSESSMENT

The Environmental Services Section (ESS) staff of CP&L reviewed the non-radiological impact of plant operation at uprated power levels on cooling water withdrawal from the Cape Fear River and discharges to the Atlantic Ocean. The proposed Brunswick Nuclear Plant (BNP) power uprate will not change the method of generating electricity nor the methods of handling cooling water from or discharges to the environment. There will be no significant increases in the environmental discharges from this uprate; therefore, no new or different types of environmental impacts are expected.

<u>Cooling Water Withdrawal</u>

The BNP uses a once-through circulating water system for dissipating heat from the main turbine condensers. This cooling system withdraws water from the Cape Fear River through a three-mile long intake canal. The heated water is discharged to the Atlantic Ocean after it travels through a six-mile long canal. A pumping station at the end of the canal pumps the water 2000 feet off of the beach through pipes. This system is operated in accordance with the requirements of a National Pollutant Discharge Elimination System (NPDES) permit, Permit No. NC0007064.

The NPDES permit currently allows the withdrawal, from the Cape Fear River, of 922 cubic feet of water per second (cfs), per unit, from December through March; 1105 cfs, per unit, April through November; and 1230 cfs through one unit only from July through September. Because the flow rate of intake circulating cooling water will not change as a result of the proposed uprated power levels, there will be no associated increase in the entrainment of planktonic organisms or impingement of fish, crabs, or shrimp.

CP&L has been monitoring the entrainment and impingement of organisms from the cooling water at BNP since 1974. A stipulation of the NPDES permits issued since 1981 is the implementation of plant modifications incorporating best available technology to reduce entrainment and impingement of organisms. To meet this stipulation, two major modifications were undertaken to exclude organisms from entering the plant, in addition to the flow minimization plan previously described. First, a diversion structure was constructed across the mouth of the intake canal in November 1982. The structure uses copper-nickel screens (9.5 cm X 15.9 cm opening) to reduce impingement by preventing large fish and shellfish from entering the intake canal. Second, to reduce the entrainment of larval organisms into the plant, fine-mesh (0.1 cm) screens were installed on on two of the four rotating screens for each unit in July 1983, and a third was installed on each unit in April 1987. Organisms impinged on these continuously rotating fine-mesh screens are washed into a one mile long flume which returns them to the estuary.

CP&L monitored the density of organisms in the Cape Fear estuary from 1976 through 1993 and concluded that plant operations did not significantly impact the relative abundance or distribution of organisms in the estuary. By letter dated August 8, 1994, to the NC Department of Environment, Health and Natural Resources, Division of Environmental Management (DEM), CP&L requested a reduction in the biological monitoring program because no significant impact had been detected after 18 years of monitoring the estuary. By letter dated August 26, 1994, with input from the NC Division of Marine Fisheries (DMF), the DEM approved the reduction of estuary monitoring and continuance of the plant impingement and entrainment monitoring. The impingement and entrainment data continue to be reported annually to the state. Because there will be no increase in cooling water withdrawal, no increase in entrainment or impingement or impacts to the estuary are expected from the power uprate. Therefore, no changes in regulatory programs are anticipated.

A by-product of the circulating water system operation is chlorine which is injected to retard the growth of biofouling organisms. The NPDES permit limits the rate of chlorine injection. The chlorine injection rate is determined by the flow rate through the circulating water system, since the circulating water system flow rate will not change as a result of operation at uprated power levels, the chlorine injection rate will not change.

Discharge Temperature

The ocean discharge mixing zone temperature limits, defined by the NPDES permit, should not be exceeded by operation at the uprated power. The uprated power has been conservatively calculated to increase the temperature of the circulating water leaving the main condensers by 1.4°F in the winter and 1.2°F in the summer (Table 6-3, GE Report). These small increases at the condenser should not significantly impact the temperature of water discharged to the ocean, after traveling more than six miles throught he discharge canal. As an example, on August 1, 1994 the ambient ocean water temperature was 83°F. With both units operating at 100% power, the water temperature at the point of ocean discharge was 91°F. At 1500 feet North and South from the point of discharge, approximately a 50 acre area, the water temperature was 83°F - ambient temperature. The NPDES permit allows an increase up to 89.5°F within an area of 1,000 acres during the summer.

Other Discharges

Effluent discharges from other systems were also reviewed for potential effects from the proposed power uprate. Effluent limits for systems such as roof drains, yard drains, low volume waste, metal cleaning waste, and the sewage treatment plant are established in the NPDES permit. Discharges from these systems are not changed by operation at uprated power, therefore, the impact on the environment from these systems is not changed.

Conclusion

Carolina Power & Light Company concludes that the proposed uprate for the Brunswick Nuclear Plant will not result in a significant adverse nonradiological environmental impact and is not an unreviewed environmental question. Because there will be no increase in the volume of water used in the once-through cooling system, there will be no power uprate-related increase in impingement or entrainment of fish/shellfish from the Cape Fear estuary. No significant change in discharge flow rates, velocities, temperatures, or chemical composition will occur. Power uprate does not impact the discharge characteristics upon which the NPDES Permit is based. The proposed increase in thermal power output of the units is the only change that will need to be reflected when the NPDES permit application is renewed in December, 1995.