UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 183 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 214 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated April 2, 1996 (BSEP 96-0123) (Reference 1), as supplemented by an earlier letter dated November 20, 1995 (BSEP 95-0535) (Reference 2), and by subsequent letters dated July 1, 1996 (BSEP 96-0242) (Reference 3), July 30, 1996 (BSEP 96-0287) (Reference 4), August 7, 1996 (BSEP 96-0300) (Reference 5), September 13, 1996 (BSEP 96-0340) (Reference 6), September 20, 1996 (BSEP 96-0348) (Reference 7), October 1, 1996 (BSEP 96-0362) (Reference 8), October 22, 1996 (BSEP 96-0392) (Reference 9), October 22, 1996 (BSEP 96-0403) (Reference 10), and October 29, 1996 (BSEP 96-0412) (Reference 20) Carolina Power & Light Company (the licensee or CP&L) submitted proposed changes to Facility Operating Licenses Nos. DPR-71 and DPR-62 and the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed changes would uprate the licensed thermal power level for each unit from the current level of 2436 megawatts thermal (MWt) to 2558 MWt. The information provided in References 3 through 10 and Reference 20 does not affect the conclusions stated in the notice of "Proposed No Significant Hazards Consideration Determination" published in the Federal Register on May 22, 1996 (61 FR 25698).

2.0 BACKGROUND

On December 28, 1990, the General Electric Company (GE) submitted GE Licensing Topical Report (LTR1) NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor (BWR) Power Uprate" (Reference 11), in which it proposed to create a generic program to increase the rated thermal power levels of BWR/4, BWR/5, and BWR/6 product lines by approximately 5 percent. The report contained a proposed outline for individual license amendment submittals and discussed the scope and depth of reviews needed and methodologies used in these reviews. In a letter dated September 30, 1991 (Reference 12), the NRC staff approved the program proposed in the GE report on the condition that individual power uprate amendment requests meet certain requirements contained in the staff's approval document.

The generic BWR power uprate program gives each licensee a consistent means to recover additional generating capacity beyond its current licensed limit, up to the reactor power level used in the original design of the nuclear steam supply system (NSSS). The original licensed power level for most licensees was based on the vendor-guaranteed power level and the design power level is often referred to as stretch power. The design power level is used in determining the specifications for all major NSSS equipment, including the emergency core cooling systems (ECCS). Therefore, increasing the rated thermal power to the design power level does not violate the design parameters of the NSSS equipment and does not significantly affect the reliability of this equipment.

The licensee's request to uprate the current licensed power level from 2436 MWt to a new limit of 2558 MWt represents approximately a 5.0 percent increase in thermal power with at least a 5 percent increase in rated steam flow. The planned approach to achieve the higher power level consists of: (1) an increase in the core thermal power with a more uniform (flatter) power distribution to create an increased steam flow, (2) a corresponding increase in feedwater flow, (3) no increase in maximum core flow, (4) a small (less than 3 percent) increase in reactor operating pressure, and (5) reactor operation primarily along equivalent rod/flow control lines. This approach is consistent with the BWR generic power uprate guidelines presented in Reference 11. The generic analyses and evaluations in NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," July 1991 and Supplements 1 & 2 (Reference 13) are based on a slightly smaller increase (4.2 percent vs. 5.0 percent) than is requested for BSEP 1 and 2. The plant-specific analysis for BSEP is presented in GE report NEDC-32466P (proprietary)/NEDO-32466 (non-proprietary), "Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant, Units 1 and 2" (Reference 14) and Supplement 1 to that report (Reference 15). The operating pressure will be increased approximately 25 pounds per square inch (psi) to assure satisfactory pressure control and pressure drop characteristics for the increased steam flow.

3.0 EVALUATION

The NRC staff reviewed the licensee's request for the BSEP, Units 1 and 2, power uprate amendments, using applicable rules, regulatory guides, sections of the Standard Review Plan (NUREG-0800), and NRC staff positions. The NRC staff also evaluated the licensee's submittal and supplements (References 1 through 10) for compliance with the generic BWR power uprate program contained in Reference 11. Individual review topics that comprise the staff's evaluation of this power uprate are discussed in detail below.

3.1 Reactor Core and Fuel Performance

3.1.1 Fuel Design and Operation

All fuel and core design limits will continue to be met by control rod pattern and/or core flow adjustments. Current design methods will not be changed for power uprate. Power uprate will increase the core power density, and will have some effects on operating flexibility, reactivity characteristics, and energy requirements. These issues are discussed in the following sections.

3.1.2 Thermal Limits Assessment

Operating limits are established to assure regulatory and/or safety limits are not exceeded for a range of postulated events as is currently the practice. The operating limit and safety limit minimum critical power ratio (MCPR) as well as the maximum average planar linear heat generation rate (MAPLHGR) and linear heat generation rate (LHGR) limits are cycle dependent and as such will be established or confirmed at each reload as is described in Reference 13.

3.1.3 Power/Flow Operating Map

The uprated power/flow operating map includes the operating domain changes for uprated power. The map includes the increased core flow (ICF) range and an uprated Extended Load Line Limit (ELLL). The maximum thermal operating power and maximum core flow correspond to the uprated power and the maximum core flow for ICF. Power has been rescaled so that uprated power is equal to 100 percent rated power. The changes to the power/flow operating map are consistent with the previously approved generic descriptions given in Reference 11.

3.1.4 Stability

The BSEP units will implement the design of Enhanced Stability Option 1-A to address the stability issue which will incorporate the power/flow map and applicable instrumentation setpoints associated with power uprate operation. Until the implementation of Option 1-A, the plant will be operated as described in Section 3.2 of Reference 13. This is acceptable to the staff.

3.1.5 Reactivity Control

3.1.5.1 Control Rod Drives (CRD) and CRD 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 CRD system was evaluated at the uprated steam flow and dome pressure.

The increase in dome pressure due to power uprate produces a corresponding increase in the bottom head pressure. Initially, rod insertion will be slower due to the high pressure. As the scram continues, the reactor pressure will eventually become the primary source of pressure to complete the scram. Hence, the higher reactor pressure will improve scram performance after the initial degradation. Therefore, an increase in the reactor pressure has little effect on scram time. The licensee has indicated that CRD performance during power uprate will meet current TS requirements. The licensee will continue to monitor by various surveillance requirements the scram time performance as required in the plant TS to ensure that the original licensing basis for the scram system is preserved.

For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit (HCU) and the vessel bottom head is

250 psi. The CRD pumps were evaluated against this requirement and were found to have sufficient capacity. The flows required for CRD cooling and driving are assured by automatic opening of the system control valve, thus compensating for the small increase in pressure. The licensee stated that the flow control valve will be adjusted, as needed, to continue to work within the optimum operating range. The licensee proposed in Reference 9 the addition of a power uprate license condition (license paragraph 2.L.(1) for Unit 1 and 2.I.(1) for Unit 2) requiring the following: if, during start-up testing for each unit immediately following implementation of the power uprate amendment on that unit, testing determines that adequate CRD cooling and drive flow is not available under uprate conditions, the licensee shall repair or modify the CRD system, as necessary, to assure that the system will continue to carry out its functions at uprated conditions. Based upon the above considerations and license condition, the NRC staff concludes that the CRD system will continue to perform all its functions at uprated power, and will function adequately during insert and withdraw modes.

3.1.5.2 Standby Liquid Control (SLCS)

The licensee evaluated the SLCS for power uprate conditions. The SLCS ability to achieve and maintain safe shutdown is not a direct function of core thermal power and, therefore, is not affected by power uprate. The SLCS capability is evaluated at each reload. The system was evaluated at the uprated pressure and found to have the ability to deliver the required flow rate for the Anticipated Transient Without Scram (ATWS) event of 86 gpm at the operating pressure increase of 25 psi. The SLCS pumps are positive displacement pumps and a small increase in pressure has no effect on the rated injection flow to the reactor. The staff agrees with the licensee's conclusion and finds that operation of the SLCS at uprated conditions is acceptable.

3.2 Reactor Coolant System and Connected Systems

3.2.1 Nuclear System Pressure Relief and Motor-Operated Valve (MOV) Capability at Increased Line and Valve Differential Pressures

The nuclear boiler pressure relief system prevents overpressurization of the nuclear system during abnormal operating transients. The plant safety/relief valves (SRVs) with reactor scram provide this protection.

The operating steam dome pressure is selected to achieve good control characteristics for the turbine control valves (TCVs) at the higher steam flow condition corresponding to uprated power. The uprate dome pressure increase will require a change in the SRV setpoints. The appropriate increase in the SRV setpoints also ensures that adequate differences between operating pressure and setpoints are maintained (i.e., the "simmer margin"), and that the increase in steam dome pressure does not result in an increase in unnecessary SRV actuation.

In Reference 20 the licensee confirmed that, for safety-related MOVs within the BSEP Generic Letter 89-10 ("Motor-Operated Valve Testing and Surveillance") program that are potentially impacted by power uprate, the increased thrust required to operate the MOVs due to expected increased line and valve differential pressures is within the capabilities of the existing valve actuators.

3.2.2 Code Overpressure Protection

The results of the overpressure protection analysis are contained in each cycle-specific reload amendment submittal. The design pressure of the reactor pressure vessel (RPV) remains at 1250 psi gauge (psig). The ASME code allowable peak pressure for the reactor vessel is 1375 psig (110 percent of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a Main Steam Isolation Valve (MSIV) closure with a failure of the valve position scram. The MSIV closure was analyzed by the licensee using the NRC-approved methods, with the following exceptions: (1) the MSIV closure event was analyzed at 102 percent of the uprated core power and 106 percent of rated steam flow; (2) the maximum initial reactor dome pressure was assumed to be 1060 psia, which is higher than the nominal uprated pressure; (3) two SRVs were assumed out-of-service for consistency with previous analyses; and (4) the analysis did not take credit for relief flow. The SRV opening pressures were +3 percent above the nominal setpoint for the available valves. The peak reactor pressure increases by 55 psi to 1344 psig, but remains below the 1375 psig ASME code limit. This overpressure analysis is acceptable to the staff.

3.2.3 Reactor Recirculation System

Power uprate will be accomplished by operating along extensions of rod lines on the power/flow map with no increase in maximum core flow. The cycle-specific core reload analyses will be performed with the most conservative core flow. The evaluation by the licensee of the reactor recirculation system performance at uprated power determined that the core flow can be maintained with no increase in pump speed. The BSEP units are licensed for ICF of 105 percent at 100 percent of current power (2436 MWt). The licensed core flow is not being increased under power uprate and the uprated power condition will be achieved with the same maximum recirculation pump speed. The licensee states that they have not experienced any pump vibration problems during ICF operation and testing for pump vibration is not necessary. However, the staff will require the licensee to monitor the recirculation system for vibration and to take corrective action should vibration occur.

The licensee estimates that the required pump head and pump flow at the uprate condition will increase the power demand of the recirculation motors and the pump net positive suction head (NPSH) but these increases are within the capability of the equipment. The cavitation protection interlock will remain the same in absolute thermal power, since it is based on the feedwater flow rate. These interlocks are based on subcooling in the external recirculation loop and thus are a function of absolute thermal power. With power uprate, slightly more subcooling occurs in the external recirculation loop due to the higher RPV dome pressure. It would therefore be possible to lower the cavitation interlock setpoint slightly, but this change would be small and is not necessary.

An evaluation by the licensee of recirculation pump NPSH found that at full power, power uprate alone does not increase NPSH required (NPSHr), and that the secondary effect of the increase in RPV pressure increases NPSH available (NPSHa), so that power uprate alone increases the NPSH margin.

The recirculation drive flow stops were reviewed by the licensee for application to uprated power conditions. Since power uprate has such a small effect on the required flow rate, the drive flow limiter continues to have adequate input and output range with the capability for low and high limit setpoints.

The licensee concluded that uprated power operation is within the capability of the recirculation system. During the startup following power uprate implementation, the licensee will perform a calibration of the core flow instrumentation at 100 percent power. The licensee has previously demonstrated adequate recirculation pump flow control up to 105 percent flow. As stated in Reference 14, these tests should also assure that no undue vibration occurs at uprate or ELLL conditions. However, the licensee has operated the recirculation pumps at the uprate conditions during ICF operation with no noticeable vibration. The licensee proposed in Reference 9 a power uprate license condition (license paragraph 2.L.(2) for Unit 1 and 2.I.(2) for Unit 2) requiring (1) monitoring of recirculation pump vibration during the initial start-up of each unit following implementation of the uprate amendment and (2) that, during this monitoring, vibration and noise shall be evaluated prior to and at uprated conditions to ensure no significant increase in vibration or noise occurs with power uprate. In Reference 9 the licensee provided further detail on how this monitoring will be performed. During power ascension above 85% reactor power, an operator will monitor the control room motor "vibration high" annunciator window associated with each recirculation pump; and, during Reactor Building tours, an operator will listen for abnormal noise at 85, 87, 90, 92, 95, 97, and 100 percent of the uprated power. Based upon the licensee's conclusions and the license condition, the NRC staff finds that adequate assurance is provided to ensure uprated power is within the capability of the recirculation system.

3.2.4 Main Steam Isolation Valves (MSIVs)

The MSIVs have been evaluated by the licensee. The MSIV operating conditions under power uprate remain within the MSIV design conditions. The BSEP units evaluation results are consistent with the bases and conclusions of the generic evaluation. Performance will be monitored by surveillance requirements in the TS to ensure original licensing basis for the MSIV's are preserved.

3.2.5 Reactor Core Isolation Cooling System (RCIC)

The reactor core isolation cooling system (RCIC) provides core cooling when the reactor pressure vessel (RPV) is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for initiation of the low pressure core cooling system. The RCIC system has been evaluated by the licensee, and is consistent with the bases and conclusions of the generic evaluation. In response to a staff request, the licensee has indicated by

Reference 3 that the recommendations of GE SIL No. 377 are not needed on the RCIC system on each BSEP unit. This recommended modification is intended to achieve the turbine speed control/system reliability desired by SIL 377, and is consistent with the requirements in the staff safety evaluation (SE) of the generic topical report. The purpose of the modification is to mitigate the concern that a slightly higher steam pressure and flow rate at the RCIC turbine inlet will challenge the system trip functions such as turbine overspeed, high steam flow isolation, low pump suction pressure and high turbine exhaust pressure. At the time the SIL was issued the licensee had modified the RCIC and has stated that the number of overspeed trips has been reduced. The licensee has provided assurance that the RCIC system will be capable of injecting its design flow rates at the higher reactor operating pressures associated with power uprate. Additionally, the licensee has stated that the reliability of the RCIC system will be monitored in accordance with criteria developed to comply with the Maintenance Rule that was implemented in accordance with 10 CFR 50.65. This is acceptable to the staff.

3.2.6 Residual Heat Removal System (RHR)

The residual heat removal system (RHR) is designed to restore and maintain the coolant inventory in the reactor vessel and to 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 these operating modes are discussed in the following paragraphs.

3.2.6.1 Shutdown Cooling Mode

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125°F in approximately 20 hours, using two RHR loops. At the uprated power level the decay heat is increased proportionally, thus slightly increasing the time required to reach the shutdown temperature to 21 hours. This increased time is judged to have an insignificant impact on plant safety.

Regulatory Guide 1.139, "Guidance for Residual Heat Removal," specifies demonstration of cold shutdown capability (200°F reactor fluid temperature) within 36 hours. For power uprate, the licensee did not perform a plant-specific BSEP evaluation for shutdown cooling based on the criteria of Regulatory Guide 1.139. However, as noted above the licensee stated that the reactor can be cooled to less than 125°F in 21 hours, which meets the 36-hour criterion. This conclusion was based on the results of other plants and the information provided above.

3.2.6.2 Suppression Pool Cooling and Containment Spray Modes

The Suppression Pool Cooling (SPC) and Containment Spray (CC) modes are designed to provide sufficient cooling to maintain the containment and suppression pool temperatures and pressures within design limits during normal operation and after a blowdown in the event of a design basis loss-of-coolant accident (LOCA). This objective is met with power uprate, since the peak suppression pool temperature analysis by the licensee (described in Section

4.1.1 of Reference 14) confirms that the pool temperature will stay below its design limit at uprated conditions. Power uprate increases the containment spray temperature by approximately 3 degrees. This 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 licensee stated that the capability of the CC mode is therefore acceptable for power uprate. The effect of higher suppression pool temperature on the NPSH of the RHR pumps during the SPC and CC modes is also discussed in Section 4.2 of Reference 14. The results show that there is adequate NPSH margin for the RHR and core spray (CS) pumps. The effect of power uprate to the above-mentioned cooling modes of the RHR system are acceptable to the staff.

3.2.7 Reactor Water Cleanup (RWCU) System

The reactor water cleanup (RWCU) system pressure and temperature will increase slightly as a result of power uprate. The licensee has evaluated the impact of these increases and has concluded that uprate will not adversely affect system integrity. The cleanup effectiveness may be diminished slightly as a result of the increased feedwater flow to the reactor; however, the current limits for reactor water chemistry will remain unchanged for power uprate. These effects on the RWCU system are acceptable to the staff.

3.3 Engineered Safety Features (ESF)

3.3.1 Emergency Core Cooling Systems (ECCS)

The effect of power uprate and the increase in RPV dome pressure on each ECCS system is addressed below. Also as discussed in the BSEP Updated Final Safety Analysis Report (UFSAR), compliance to the NPSH requirements of the ECCS pumps is based on a zero containment pressure and the maximum expected temperature of pumped fluids. Based on this temperature and no credit for containment pressure, it was found that there is adequate NPSH available to the RHR and CS pumps. Therefore, power uprate will not affect compliance to the ECCS pump NPSH requirements.

3.3.1.1 High Pressure Core Injection System (HPCI)

The HPCI system has been evaluated by the licensee, and is consistent with the bases and conclusions of the generic evaluation. The licensee has indicated that they have implemented the guidance contained in GE SIL 480 on the HPCI system for each unit. The licensee stated that the HPCI system is capable of delivering its design flow at the uprate conditions. The licensee also plans to perform startup testing on HPCI during the initial startup after being licensed at uprated power. The licensee stated that the HPCI system is capable of injecting the design flow rates at the higher reactor operating pressures associated with power uprate. The licensee stated in Reference 3 that the reliability of the HPCI system will be monitored in accordance with the criteria developed to comply with the Maintenance Rule that was implemented in accordance with 10 CFR 50.65. This is acceptable.

3.3.1.2 Low Pressure Core Injection System (LPCI mode of RHR)

The hardware for the low pressure portions of the RHR are not affected by power uprate. The upper limit of the low pressure ECCS injection setpoints will not be changed for power uprate, therefore the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. In addition, the RHR system shutdown cooling mode flow rates and operating pressures will not be increased. Therefore, since the system does not experience different operating conditions due to power uprate, there is no impact due to power uprate. The licensee stated that the BSEP units are bounded by the generic analyses presented in Section 4.1 of Reference 13. The NRC staff finds the licensee's conclusion acceptable.

3.3.1.3 Core Spray System (CS)

The hardware for the low pressure core spray are not affected by power uprate. The upper limit of the low pressure ECCS injection setpoints will not be changed for power uprate, therefore the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. Therefore, since these systems do not experience different operating conditions due to power uprate, there is no impact due to power uprate. Also, the impact of power uprate on the long-term response to a LOCA will continue to be bounded by the short-term response. The licensee stated that the BSEP units are bounded by the generic analyses presented in Section 4.1 of Reference 13. The NRC staff finds the licensee's conclusion acceptable.

3.3.1.4 Automatic Depressurization Systems (ADS)

The automatic depressurization system (ADS) uses safety/relief valves to reduce reactor pressure following a small break LOCA with HPCI failure. This function allows LPCI and CS to flow to the vessel. The ADS initiation logic and ADS valve control are adequate for uprate. Plant design requires a minimum flow capacity for the SRVs, and that ADS initiate after a time delay on either low water level plus high drywell pressure, or on low water level alone. The ability to perform either of these functions is not affected by power uprate.

3.3.2 ECCS Performance Evaluation

The 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 must satisfy the requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K. The fuel, used in BSEP Units 1 and 2, was analyzed by the licensee with the NRC-approved methods. The results of the base ECCS-LOCA analysis using NRC-approved methods is presented in NEDC-31624P (Reference 16), the plant-specific ECCS-LOCA results for the BSEP units.

The licensee used the NRC staff-approved SAFER/GESTR (S/G) methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria. The S/G-

LOCA analysis for the BSEP Units 1 and 2 was performed by the licensee with the appropriate reload fuel in accordance with NRC requirements to demonstrate conformance with the ECCS acceptance criteria of 10 CFR 50.46 and Appendix K in Supplement 3 (Reference 17). The base S/G-LOCA (Reference 16) and the Reference 17 analyses were performed at a nominal power level of 2680 MWt (110 percent of the current rated power of 2436 MWt) and an Appendix K power level of 2733 MWt (102 percent of 2680 MWt) in anticipation of future power uprate. Therefore, this analysis bounds the requested power uprate of 2558 MWt. The licensee also provided assurance that the break spectrum analyses and the single failure analysis documented in the base report (Reference 16) remain applicable to the planned power uprate analysis to 2558 MWt.

Single loop operation (SLO) is not currently licensed for the BSEP units, and is not being requested under this power uprate license amendment request. The licensee will submit any MCPR or MAPLHGR multipliers with the core operating limits report (COLR) as is the usual practice.

The licensee provided assurance in Reference 3 that the power uprate and fuel reload will not change the limiting break, single failure, or the break spectrum as compared to the existing analysis. Therefore, the staff finds that BSEP Units 1 and 2 will continue to meet the NRC LOCA licensing analysis and results requirements. The licensee must submit the results of the plant-specific LOCA analysis at each reload.

3.4 Reactor Safety Performance Features

3.4.1 Reactor Transients

Reload licensing analyses evaluate the limiting plant transients. Disturbances of the plant caused by a malfunction, a single failure of equipment, or personnel error are investigated according to the type of initiating event. The licensee will use its NRC-approved licensing analysis methodology to calculate the effects of the limiting reactor transients as identified in the generic guidelines. The limiting events for the BSEP units were identified as those analyzed in References 14 and 15. The relatively small changes in rated power and maximum allowed core flow are not expected to affect the selection of limiting events. The events explicitly evaluated for the power uprate analysis are:

1. Loss of Feedwater Heating (LOFWH)
2. Feedwater Controller Failure (FWCF)
3. Generator Load Rejection without Bypass (GLRWOB)
4. Turbine Trip without Bypass (TTWOB)
5. Rod Withdrawal Error (RWE)
6. Slow Recirculation Flow Increase
7. Inadvertent HPCI Actuation
8. Loss of Feedwater Flow (LOFW)

The limiting events which establish the MCPR operating limits are currently GLRWOB, FWCF, and LOFWH. These events are also limiting at power uprate conditions. The analyses were performed by the licensee up to a maximum power level of 102 percent of the uprated power level, or 2558 MWt to account for power uncertainty, and at 105 percent steam flow. The input parameters for the transient analyses are presented in Table 9-1 of Reference 14, and the results of the transient analyses are presented in Table 9-2 of Reference 14. The analysis SRV setpoints used are shown in Table 5-1 of Reference 14, with the tolerance increased to 3 percent. The Unit 1 analysis was used as the representative fuel cycle for power uprate because it also bounds Unit 2. The power uprate analysis used the staff-approved GEMINI methodology. Direct or statistical allowance for 2 percent power uncertainty is included in the analysis. Most of the transients events are analyzed at the full uprated power and maximum allowed core flow operating point which bounds the power/flow map shown in Figure 2-1 of Reference 14.

Cycle-specific analyses will be done at each reload and the results will be part of the COLR developed by the licensee. The safety limit minimum critical power ratio (SLMCPR) is calculated by the licensee as part of the reload licensing analyses using the NRC-approved methodology for the appropriate reload fuel. No change will be made to this methodology due to power uprate or increased core flow. The analysis plan proposed by the licensee is acceptable. The licensee will submit the results of the cycle-specific analysis with each reload document. The NRC staff finds that use of NRC-approved methodology as described above will ensure that the effects of transients will be within applicable design and safety limits.

3.4.2 Special Events

3.4.2.1 Anticipated Transients Without Scram (ATWS)

A generic evaluation of the ATWS event is presented in Section 3.7 of Supplement 1 to Reference 17. This evaluation concludes that the ATWS acceptance criteria for fuel, RPV, and the containment integrity will not be violated for power uprate if the following are met: reactor power increase is equal to or less than 5 percent; dome pressure increase is equal to or less than 40 psi; SRV opening setpoint increase is equal to or less than 80 psi; and ATWS high pressure setpoint increases equal to or less than 20 psi. The BSEP power uprate meets the four criteria except that the ATWS high pressure setpoint will be increased by 50 psi. The licensee evaluated the limiting ATWS event, the MSIV closure. The RPV integrity was reanalyzed with the power uprate input parameters of 2558 MWt; reactor dome pressure of 1048 psia; SRV opening setpoints increased by 25 psi, with tolerance increased to 3 percent; ATWS high pressure analytical limit increased by 50 psi to 1170 psig; and with one SRV assumed out-of-service (the SRV with the lowest setpoint). The results showed the peak RPV pressure to be 1493 psig, which is below the American Society of Mechanical Engineers (ASME) Code limit of 1500 psig. The licensee stated that based on the generic analysis in Reference 17 and the plant-specific BSEP analysis, power uprate will not result in any ATWS acceptance criteria being exceeded. The licensee stated in Reference 4 that the other acceptance criteria specified in the generic report are unaffected by the increase in ATWS-RPT analytical limit. This analysis is acceptable to the staff.

3.4.2.2 Station Blackout (SBO)

Plant response and coping capabilities for a station blackout event are impacted by operation at the uprated power level due to the increase in the operating temperature of the primary coolant system, increase in 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 coping time changed.

The licensee stated that the BSEP response to a postulated SBO is to use the HPCI system with Direct Current (DC) augmentation from the other unit to cope for 4 hours, and that the emergency diesel-generator and station battery performance are adequate for response to an SBO with power operation.

The following areas contain equipment necessary to mitigate the SBO event: Control Room, Diesel-Generator Basement and 480 V EBUS Switchgear Rooms; RCIC and HPCI Equipment Room; ECCS Pipe Tunnel; and Containment.

The temperature increases in the Control Room, Diesel-Generator Basement and 480 V emergency bus (EBUS) Switchgear Rooms are not affected by power uprate. The RCIC and HPCI equipment room temperatures and ECCS pipe tunnel temperature will increase. However, the licensee states that 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 the containment pressure about 2 psi, but the increases are small enough to not affect equipment availability. The condensate water requirement increases less than 3 percent; however, the current condensate storage tank design assures that adequate water volume is available. Based on the above evaluation and assurances by the licensee, the SBO coping capabilities are not adversely affected by power uprate and are acceptable.

The limiting parameters for SBO events lasting longer than 4 hours are water inventory for decay heat removal, class 1E battery capacity, compressed air capacity, and the effects of loss of ventilation. Power uprate will result in more decay heat which will require a slightly larger water inventory. However, the current SBO analysis provides for adequate water inventory to meet the additional requirements of power uprate.

Class 1E battery capacity and the compressed air system are unaffected by power uprate, and power uprate will not increase demand on these systems for SBO scenarios. The capacity of these systems will, therefore, remain adequate.

The licensee's SBO analysis is acceptable to the NRC staff. Based on NRC staff review of the licensee's rationale and the experience gained from NRC review of power uprate applications for similar BWR plants, the NRC staff finds that the impact on the coping of an SBO event due to plant operations at the proposed uprated power level will be insignificant.

3.5 Containment System Performance

The Brunswick Nuclear Plant, Units 1 & 2, UFSARs provide the analyses of the containment response to various postulated accidents that constitute the containment design basis. Operation with the power uprate would change certain conditions of the analyses. Section 5.10.2 of Reference 11 requires the power uprate applicant to show that the following analyses remain acceptable under uprated power conditions: (1) containment pressure and temperature; (2) LOCA containment dynamic loads; and (3) safety-relief valve dynamic loads. Appendix G of Reference 11 prescribes the approach to be used by applicants for performing required plant-specific analyses.

Appendix G of Reference 11 states that the applicant will analyze short-term containment response using the M3CPT code. M3CPT is used to analyze the period from when the break begins to when pool cooling begins, and determines the containment pressure and temperature response (Section 4.1.1 of Ref. 14), containment dynamic loads, (Section 4.1.2 of Ref. 14) and data related to equipment qualification analyses (Section 10.2 of Refs. 14 and 15). The NUREG-0661, "Mark I Containment Long Term Program Safety Evaluation" (Reference 18), approach was applied on a plant-specific basis to Brunswick and the approach was approved by the staff.

Appendix G of Reference 11 also requires the applicant to perform long-term containment heatup (suppression pool temperature) analyses for the limiting UFSAR events to show that the pool temperatures will remain within limits for:

Containment design temperature
Net positive suction head (NPSH)
Pump seals, piping design temperature, and other applicable limits

These analyses use the SHEX code. The staff approved the use of SHEX (and ANS (American Nuclear Society) 5.1 decay heat model) on a plant-specific basis, provided that confirmatory calculations for validation of the results were included in the plant-specific request (Letter from NRC to Gary L. Sozzi of GE, July 13, 1993). SHEX is partially based on M3CPT and is used to analyze the period from when the break begins until after peak pool heatup (i.e. the long-term response).

The results of the licensee's analyses are discussed in the following sections.

3.5.1 Containment Pressure and Temperature Response

Short-term and long-term analyses of the containment pressure and temperature response following a large break inside the drywell are documented in the Brunswick UFSARs. The short-term analysis is performed to determine the peak drywell pressure during the initial blowdown of the reactor vessel inventory into containment following a large break inside the drywell (Design Basis Accident (DBA) LOCA), while the long-term analysis is performed to determine the peak pool temperature response considering decay heat addition.

3.5.1.1 Differences in Computer Codes and Input Parameters

The short- and long-term containment pressure and temperature response given in the UFSAR were calculated using the CONTEMPT-PS computer code, while the present short- and long-term analyses use M3CPT and SHEX, respectively. These codes (SHEX and M3CPT) were also used to perform the Brunswick specific analysis for long- and short-term containment response under the Mark I program.

Key differences between the input parameters used in the UFSAR analyses and analyses conducted for the power uprate are given below. The licensee indicated that most of the differences in the input parameters (not including the reactor conditions associated with the power uprate) were established during the Mark 1 Containment Long Term Progam (LTP).

- Initial drywell temperature is raised from 130°F to 135°F.

- Initial suppression pool and suppression chamber airspace temperatures are raised from 90°F to 95°F.

- Initial drywell and wetwell pressure in the power uprate analysis are assumed equal at 2.5 psig (corresponding to a 0 differential pressure consistent with the Mark 1 Containment LTP), versus a drywell and wetwell pressure of 1.3 psig and 0.0 psig, respectively, used in the UFSAR analysis.

- Minimum and maximum wetwell volumes are changed to 122,000 cu. ft. and 124,000 cu. ft, respectively, from the UFSAR values of 124,000 cu. ft. and 134,600 cu. ft., respectively. No change was made in the minimum suppression pool water volume.

- The overall vent loss coefficient is increased to 5.17 from the UFSAR value of 4.06.

- The ANS/ANSI 5.1 decay heat model is used with a 2-sigma uncertainty.

- For the current long-term analysis, all feedwater at high energy was added to the vessel to maximize the suppression pool temperature response. This was not typically included during the time of the original UFSAR analysis.

The licensee also provided a comparison between the peak drywell pressure and peak bulk suppression pool temperature given in the UFSAR analysis and calculated with CONTEMPT-PS, and that calculated using the M3CPT code (used to calculate peak pressure) and SHEX (used to calculate the peak bulk pool temperature). The comparisons were done using current rated power. The peak drywell pressure was calculated to be 36.8 psig versus an FSAR value of 49.4 psig, and the calculated peak bulk pool temperature was 197°F versus the UFSAR value of 205°F. The difference is due to the GE SHEX and M3CPT computer codes used, more realistic decay heat model and updated assumptions. The licensee provided details in Reference 6.

3.5.1.2 Long-Term Suppression Pool Temperature Response

(1) Bulk Pool Temperature

The licensee indicates that the long-term bulk suppression pool temperature response was evaluated for the DBA LOCA for both 102 percent of original rated power and 102 percent of uprated power using the SHEX code and the more realistic decay heat model (ANS/ANSI 5.1 with a 2-sigma uncertainty) than used in the current UFSAR. The licensee also used updated initial drywell and wetwell temperatures, updated initial pressure, updated wetwell volume, updated overall vent loss coefficient and feedwater assumptions as indicated in 3.5.1.1.

The analysis shows that, using current methodology, the power uprate increases the peak pool temperature by 4°F, resulting in a DBA LOCA peak suppression pool temperature of 201°F. This is below the design temperature of 220°F.

The licensee also presented calculations that show the available NPSH for CS and RHR pumps remains adequate for both units during the long-term cooling period following a DBA-LOCA. The revised NPSH calculations do not take credit for containment pressurization. See Section 3.3 for details.

Based on the results of these analyses, the staff concludes that the peak bulk suppression pool temperature response remains acceptable from both NPSH and structural design standpoints after the power uprate.

(2) Local Pool Temperature with SRV Discharge

Because of concerns resulting from unstable condensation observed at high pool temperatures in plants without quenchers, a local pool temperature limit for SRV discharge is specified in NUREG-0783, "Suppression Pool Temperature Limit for BWR Containments." The licensee indicated that since both units of Brunswick have quenchers, no evaluation of this limit is considered necessary. Elimination of this limit for plants with quenchers on the SRV discharge lines is justified in GE report NEDO-30832, "Elimination of Limits on Local Suppression Pool Temperature for SRV Discharge with Quenchers."

Based on the above, the staff concludes that the local pool temperature limit will remain acceptable after the power uprate.

3.5.1.3 Containment Gas Temperature Response

The licensee indicates that the drywell design temperature has been determined based on a bounding analysis of the superheated gas temperature reached as a result of steam blowdown to the drywell during a LOCA. The bounding analysis is performed for a worst-case scenario based on a reactor pressure that is independent of power level. The licensee's analysis at uprated power results in a peak drywell temperature of 288°F, which is below the drywell design temperature of 300°F as indicated in the UFSAR.

The licensee indicates that the wetwell gas space peak temperature was calculated assuming thermal equilibrium between the pool and the gas space.

Using updated methods, the re-analysis shows that the maximum bulk pool temperature and wetwell temperature would increase slightly (approximately 4°F) after a LOCA to 201°F, but for both units would remain below the original UFSAR value of 205°F and below the wetwell design temperature of 220°F.

Based on its review, the staff concludes that the containment drywell and wetwell gas temperature response will remain acceptable after the power uprate.

3.5.1.4 Short Term Containment Pressure Response

The licensee's submittal contains a short-term containment response analysis performed to demonstrate that operation with uprated power would not result in exceeding the containment design pressures. The analysis was performed for a double-ended guillotine break of a recirculation suction line, the most limiting DBA LOCA for containment pressure for Brunswick, and covers the blowdown period. During this period, the maximum drywell pressure and maximum differential pressure between the drywell and wetwell occur. The GE M3CPT computer code was used and 102 percent of the uprated power level assumed. The re-analysis predicts a maximum containment pressure of 40.9 psig, which is bounded by the UFSAR original analysis value of 49.4 psig and remains below the containment design pressure of 62 psig for both units.

Based on its review, the staff concludes that the containment pressure response following a postulated LOCA will remain acceptable after the power uprate.

3.5.2 Containment Dynamic Loads

3.5.2.1 LOCA Containment Dynamic Loads

Reference 11 requires that the power uprate applicant determine if the containment pressure, suppression pool temperature and vent flow conditions calculated with the M3CPT code for the power uprate are bounded by the analytical or experimental conditions on which the previously analyzed LOCA dynamic loads are based. If the new conditions are within the range of conditions used to define the loads, then LOCA dynamic loads are not affected by the power uprate and thus do not require further analysis.

The LOCA containment dynamic loads considered for the power uprate are based on the short-term LOCA analyses, which provide calculated values for the controlling parameters for the dynamic loads throughout the blowdown. The key parameters are the drywell and wetwell pressure, the vent flow rates, and the suppression pool temperature. The LOCA dynamic loads considered in the power uprate evaluation include pool swell, condensation oscillation (CO), chugging, and vent thrust loads.

The licensee states that the short-term containment response conditions with uprated power are within the range of test conditions used to define the pool swell and condensation oscillation loads for the plant. Furthermore, the long-term response conditions with the power uprate for which chugging would occur are within the conditions used to define the chugging loads, and the vent thrust loads are calculated to be less than plant-specific values determined during the Mark I Containment LTP. Therefore, the LOCA dynamic loads for Brunswick Units 1 and 2 would not be impacted by the uprate.

Based on the above, the staff concludes that the LOCA containment dynamic loads will remain acceptable after the power uprate.

3.5.2.2 Safety-Relief Valve (SRV) Containment Dynamic Loads

The SRV containment dynamic loads comprise discharge line loads (SRVDL), suppression pool boundary pressure loads, and drag loads on submerged structures. The loads are influenced by the SRV opening setpoint pressure, the initial water leg height in the SRVDL, SRVDL geometry, and suppression pool geometry.

For the first SRV actuations following an event, the only parametric change introduced by the power uprate which can affect the SRV loads is the increase in SRV opening setpoint pressure. An increase in the setpoint pressure will result in higher SRV flow rates and, therefore, higher SRV loads. Subsequent actuation loads may be affected by changes in the SRV discharge line water level at the time of subsequent actuations - a higher water level will result in higher SRV loads. Reference 11 states that if the SRV setpoints are increased, the power uprate applicant should attempt to show that there is sufficient margin in the SRV design loads to accommodate the higher setpoints.

The licensee is increasing the analytical limits for the setpoints by approximately 4.3 percent because of the power uprate. However, the SRV flow rates used in the Plant Unique Analysis Report (PUAR) are conservative relative to the values calculated with the current analytical setpoint and to certified values. The maximum increase in the SRV flow rates with uprated power relative to PUAR values is only 0.1 percent, which would increase the SRV loads by approximately 0.1 percent. This increase is considered negligible relative to the conservatism in the SRV loads defined as part of the Mark 1 LTP for Brunswick. Therefore, the power uprate would not impact the SRV load definition for the first SRV actuations.

The licensee indicates that in the PUAR, the water leg prior to SRV opening used to define the subsequent actuation loads conservatively assumes the maximum calculated SRVDL reflood height. The maximum SRVDL reflood height is a function of the SRVDL vacuum breaker size and SRVDL geometry, which would not be impacted by the power uprate. Therefore, the power uprate has no effect on the water leg prior to SRV opening, and no impact on the subsequent actuation loads. The loads would remain below their original design values.

Based on the above review, the staff concludes that the SRV containment dynamic loads will remain acceptable after the power uprate.

3.5.2.3 Subcompartment Pressurization

Reference 11 methodology does not require subcompartment reanalysis.

However, the licensee indicates that due to operation at a higher pressure with the power uprate, the asymmetrical loads on the vessel, attached piping, and biological shield wall resulting from a postulated pipe break in the annulus between the reactor vessel and the shield wall would increase by approximately 2 percent. The biological shield wall and component designs remain adequate because of the large degree of conservatism in the originally analyzed loads, which bound the loads which would result from uprated conditions by approximately 50 percent.

Based on the above, the staff concludes that the subcompartment pressurization effects will remain acceptable after the power uprate.

3.5.3 Containment Isolation

The licensee indicates that the system designs for containment isolation would not be affected by the power uprate. The capability of the actuation devices to perform with the higher pressure and flow has been evaluated and determined to be acceptable. Motor-operated containment isolation valves required to reposition in the event of an accident will, under uprated conditions, comply with Generic Letter (GL) 89-10.

Based on its review, the staff finds that the operation of the plant at the uprated power level will not impact the containment isolation system.

3.5.4 Post-LOCA Combustible Gas Control

Under LOCA conditions, combustible gases would be generated in containment from the radiolysis of water (generating oxygen and hydrogen) and the metal-water reaction with the fuel cladding (generating hydrogen).

The Containment Atmospheric Control (CAC) system is used to maintain the primary oxygen concentration below flammability limits (i.e. below a 5 percent oxygen/4 percent hydrogen mixture as given in Regulatory Guide 1.7). This function is provided by inerting the containment with nitrogen prior to startup with the Containment Inerting subsystem, maintaining an inert atmosphere via nitrogen makeup during operation with the Atmospheric Makeup subsystem, and controlling the post-LOCA containment oxygen concentration below 5 percent with the Containment Atmosphere Dilution (CAD) subsystem. Because they are considered non-purge/repressurization plants with respect to hydrogen as classified in GL 84-09, the Brunswick units do not have nor are they required to have hydrogen recombiners for post-LOCA hydrogen control.

The CAD system is capable of maintaining the containment atmosphere below 5 percent oxygen by volume. A power uprate will cause an increase of radiolytic oxygen proportional to the increase in plant thermal power. Based on the 5 percent increase in core power due to the uprate, it is expected that the CAD system will need to be initiated slightly sooner. Because CAD initiation time is the basis for the limiting pressure, a slightly earlier initiation means that the limiting containment pressure of 31 psig (one-half of design pressure) would be reached sooner. The licensee indicates that the initiating time will still be beyond the regulatory minimum of 30 minutes and the limiting pressure will be reached beyond 30 days. These results remain bounded by the Generic Evaluation performed in NEDO-22155, "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark 1 Containments," June 1982, which has been reviewed and approved by the NRC staff.

Based on its review, the staff concludes that control of post-LOCA combustible gases will remain acceptable after the power uprate.

3.6 Residual Heat Removal (RHR) System - Fuel Pool Cooling Assist Mode

The impact of plant operations at the proposed uprated power level on the RHR fuel pool cooling assist mode is addressed in Section 3.8.

3.7 Other Engineered Safety Features (ESF) and Power Distribution Systems

3.7.1 Emergency Cooling Water System

The impact of plant operations at the proposed uprated power level on safety-related and nonsafety-related water systems are addressed in Section 3.9.

3.7.2 Emergency Core Cooling Auxiliary Systems

The impact of plant operations at the proposed uprated power level on power dependent HVAC and other auxiliary systems are addressed in Sections 3.10 and 3.12.

3.7.3 Main Control Room Atmosphere Control System (CRACS)

The control room atmosphere control system containing an emergency filtration system is designed to maintain the control room envelope at a slightly positive pressure relative to the outside atmosphere and thus minimize unfiltered inleakage of contaminated outside air into the control room following a LOCA. Plant operations at the proposed uprated power level do not change the design and operational aspects of the CRACS. The licensee performed evaluations of the effects of plant operations at the proposed uprated power level on the CRACS and demonstrated that the control room radiological consequences resulting from plant operations at the proposed uprated power level remain within the control room atmosphere design limits.

Based on NRC staff review and the experience gained from review of power uprate applications for similar BWR plants, the NRC staff agrees with the licensee's evaluations that plant operations at the proposed uprated power level will have an insignificant or no impact on the CRACS.

3.7.4 Offsite and Onsite Power Distribution Systems/Grid Stability

In Sections 6.1.1 and 6.1.2 of Reference 2 and in Reference 10 the licensee described its evaluation for the power uprate regarding continued conformance with General Design Criterion (GDC) 17 of Appendix A to 10 CFR Part 50. GDC 17 addresses onsite and offsite electrical supply systems for the electrical equipment important to safety. The evaluation included consideration of both the onsite and offsite power systems.

The NRC staff has concluded that the licensee has evaluated the impact of the power uprate on the necessary electrical systems and components and determined that the safety functions of the electrical power system can be maintained. The NRC staff finds that the power uprate is acceptable because conformance to GDC 17 is maintained.

3.8 Fuel Pool Cooling Systems

The primary fuel pool cooling system (PFPCS) is designed to remove the decay heat released from the stored spent fuel assemblies and maintain a pool water temperature at or below design temperature under normal operating conditions for heat loads up to those anticipated to result from a recent partial core off-load. Supplemental fuel pool cooling may be provided by the RHR system or the supplemental spent fuel pool cooling system (SSFPCS) in the event of a full core off-load, which has been a regular refueling practice at Brunswick.

As a result of plant operations at the proposed uprated power level, the decay heat load for any specific fuel discharge scenario will increase slightly. The licensee performed evaluations and concluded that plant operations at the proposed uprated power level will not have any negative effect on the capability of the PFPCS or the fuel pool cooling assist mode of the RHR system. Power uprate also will not impact the heat removal capability of the SSFPCS. However, the higher decay heat rate resulting from power uprate could cause a delay in removing the RHR system from service during refueling. The current design basis specifies that the SSFPCS, assuming a single failure and operating in parallel with one loop of the PFPCS, can maintain the spent fuel pool temperature at or below 125°F. Until that condition is satisfied, the fuel pool cooling assist mode of RHR must be available in order to meet the design basis.

The licensee stated that their approach to decay heat removal during refueling outages has been to evaluate the decay heat loads and the decay heat removal systems available for each refueling outage. Both bounding and outage-specific analyses have been utilized for various refueling sequences. As proposed by the licensee in Reference 9, a condition will be added to the license for each unit (license paragraph 2.L.(3) for Unit 1 and 2.I.(3) for Unit 2) requiring that the decay heat loads and the decay heat removal systems available for each refueling outage be evaluated, and bounding or outage specific analyses be used for various refueling sequences. Where a bounding engineering evaluation is in place, a refueling-specific assessment shall be made to ensure that the bounding case encompasses the specific refueling sequence. In both cases (i.e., bounding or outage-specific evaluations), compliance with design basis assumptions shall be verified.

On the basis of the license condition to evaluate decay heat loads relative to available decay heat removal capability to verify compliance with design basis assumptions, the NRC staff finds that plant operations at the proposed uprated power level are acceptable with respect to decay heat removal from the spent fuel pool.

Separately, an issue associated with spent fuel pool cooling adequacy was identified in NRC Information Notice 93-83 and its Supplement 1, "Potential

Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," dated October 7, 1993 and August 24, 1995, respectively, and in a 10 CFR Part 21 notification, dated November 27, 1992. The staff is evaluating this issue, as well as broader issues associated with spent fuel storage safety, as part of the NRC generic issue evaluation process. If the generic review concludes that additional requirements in the area of spent fuel pool safety are warranted, the staff will address those requirements to the license under separate cover.

3.9 Water Systems

3.9.1 Service Water Systems

The service water systems are designed to provide cooling water to various systems (both safety-related and nonsafety-related).

3.9.1.1 Safety-Related Loads

These safety-related loads include the loads from the emergency equipment service water system and the RHR service water system. All heat removed by these systems is rejected to the atmosphere via the ultimate heat sink.

3.9.1.1.1 Emergency Equipment Service Water (EESW) System

The EESW system removes heat from diesel generator coolers, ECCS room coolers and RHR pump seal coolers following a LOCA. The licensee stated that the cooling loads for the EESW system remain virtually the same as that for the current rated power level operation because the diesel generator loads, ECCS room cooler loads, and the RHR System flows remain unchanged for LOCA conditions following plant operations at the proposed uprated power level.

Based on our review and the experience gained from our review of power uprate applications for similar BWR plants, the NRC staff finds that plant operations at the proposed uprated power level will have an insignificant or no impact on the EESW system.

3.9.1.1.2 Residual Heat Removal Service Water (RHRSW) System

The residual heat removal service water system provides safety-related cooling water to the RHR system under normal or post-accident conditions. The licensee stated that in the re-analysis for containment pressure and temperature response to demonstrate the containment system capability to operate with uprated power, the RHR cooling capacity during post-LOCA was assumed not to increase for power uprated conditions. Therefore, power uprate will not change the cooling requirements on RHR and its associated service water system for post-LOCA conditions. During shutdown cooling with the RHR, heat loads on the RHRSW system will increase proportionally to the increase in reactor operating power level. The licensee performed evaluations and stated that the existing design cooling capacity of the RHRSWS is adequate for the proposed uprated power operation.

Based on NRC staff review of the licensee's rationale and the experience gained from review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the RHRSW system.

3.9.1.2 Nonsafety-Related Loads

The licensee stated that 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 bus cooler heat loads. The increase in service water heat loads from these sources due to uprated operation is approximately proportional to the power uprate. The licensee performed evaluations which demonstrate that the service water system is adequate for power uprate conditions.

Based upon NRC staff experience with previous power uprate applications for similar BWR plants, the licensee's evaluations demonstrating that the service water system is adequate for power uprate conditions, and the fact that these are nonsafety-related loads, the NRC staff has not reviewed the impact of the increased nonsafety-related loads under the proposed uprated power operations.

3.9.2 Main Condenser, Circulating Water, and Normal Heat Sink Systems

The main condenser, circulating, and normal heat sink systems are designed to provide the main condenser with a continuous supply of cooling water for removing heat rejected to the condenser by turbine exhaust, turbine bypass steam, and other exhausts over the full range of operating loads thereby maintaining low condenser pressure. The licensee stated that the performance of the main condenser, circulating water, and normal heat sink systems was evaluated and found adequate for plant operations at the proposed uprated power level.

Based upon (1) NRC staff experience with previous uprate applications for similar BWR plants, (2) the licensee's conclusion that the performance of the main condenser, circulating water, and normal heat sink systems will be adequate at the proposed uprated power level, and (3) the fact that the main condenser, circulating water, and normal heat sink systems do not perform any safety-related function, the NRC staff has not reviewed the impact of the proposed uprated power operations on the design and performance of these systems.

3.9.2.1 Discharge Limits

The licensee compared the state discharge limits to current discharges and bounding analysis discharges for power uprate. The comparison demonstrates that the plant will remain within the state discharge limits during operations at uprated power level.

Based on this comparison and NRC staff experience gained from review of power uprate applications for similar BWR plants, the NRC staff finds that plant operations at the proposed uprated power level will have an insignificant impact on the main condenser/circulating water discharge limits.

3.9.3 Reactor Building Closed Cooling Water (RBCCW) System

The reactor building closed cooling water system is designed to remove heat from various auxiliary plant equipment housed in the reactor building. The licensee performed evaluations and stated that the increase in heat loads to this system due to uprated power operations is insignificant and is within the existing design heat loads.

Based on NRC staff review of the licensee's rationale and the experience gained from review of power uprate applications for similar BWR plants, the NRC staff finds that plant operations at the proposed uprated power level do not change the design aspects and operations of the RBCCW system. Therefore, the NRC staff finds that the impact of plant operations at the proposed uprated power level on the RBCCW system is insignificant.

3.9.4 Turbine Building Closed Cooling Water (TBCCW) System

The heat loads on the turbine building closed cooling water system which are power dependent and are increased by power uprate are those related to the operation of the turbine-generator. The remaining TBCCW system heat loads are not strongly dependent upon reactor power and will not increase significantly. The licensee performed evaluations and stated that the increase in heat loads to this system due to uprated power operations is insignificant and is within the existing design heat loads.

Based upon NRC staff experience with previous reviews of power uprate applications for similar BWR plants, the licensee's conclusion that the increase in heat loads to this system due to uprated power operations is insignificant and within design heat loads, and the fact that the TBCCW system does not perform any safety-related function, the NRC staff has not reviewed the impact of the proposed uprated power operations on the design and performance of this system.

3.9.5 Ultimate Heat Sink (UHS)

The UHS for Brunswick 1/2 is the Cape Fear River. The cooling water intake for the Brunswick plant is located on the river approximately 5 miles from where the river enters the Atlantic Ocean. The Brunswick plant discharges into the Atlantic Ocean. The licensee stated that the temperature of the river is unaffected by uprate. Discharge temperatures to the Atlantic Ocean may increase in proportion to the 5 percent increase in decay heat; however, the impact on the Atlantic Ocean due to the increase in temperature is negligible. In addition, the licensee stated that an evaluation of plant operating parameters impacted by the power level uprate concludes that no significant environmental impact will result from plant operations at the proposed uprated power level.

Based on NRC staff review, the NRC staff finds that plant operations at the proposed uprated power level will have an insignificant impact on the UHS.

3.10 Power Dependent Heating, Ventilation, and Air Conditioning (HVAC) Systems

The HVAC systems consists mainly of cooling supply, exhaust and recirculation units in the reactor building, drywell, and turbine building. The licensee stated that the areas affected by power uprate consist of the drywell, steam tunnel, and feedwater heater and condenser areas in the turbine building. The licensee performed evaluations which indicated that the area design temperatures for all plant operating modes envelop the temperatures resulting from the anticipated increase in heat loads due to the plant operations at the proposed uprated power level. Thus, the existing design of the HVAC systems for the above cited areas is acceptable for plant operations at the uprated power level.

Based on NRC staff review of the licensee's rationale and the experience gained from our review of power uprate applications for similar BWR plants, the NRC staff finds that plant operations at the proposed uprated power level has an insignificant or no impact on the HVAC systems for the above cited areas.

3.11 Fire Protection

The licensee stated that Brunswick calculations associated with 10 CFR Part 50, Appendix R were reviewed. It was concluded that plant operations at the proposed uprated power level does not affect the ability of the Appendix R systems to perform their safe shutdown function.

Fire suppression or detection is not expected to be impacted due to plant operations at the proposed uprated power level since there are no physical plant configurations or combustible load changes resulting from the uprated power operations. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change and are acceptable for the uprated conditions, and the operator actions required to mitigate the consequences of a fire are not affected.

Based on NRC staff review of the licensee's rationale and the experience gained from review of power uprate applications for similar BWR plants, the NRC staff finds that the fire suppression and detection systems are not power dependent and will not be affected by plant operations at the proposed uprated power level.

3.12 Other Systems Reviewed for Impact By Power Uprate

The licensee identified other systems which are not affected by plant operations at the proposed uprated power level. The NRC staff reviewed the following systems for which it has review responsibility:

- Auxiliary Boiler
- Condensate Makeup
- Screen Wash
- Turbine-generator Lube Oil
- Diesel Generator and its associated supporting systems

- Instrument Air
- Service Air
- Fire Detection

Based on NRC staff review, the NRC staff finds that plant operation at the proposed uprated power level has no impact on these systems.

3.13 Systems With Minimal Impact

The licensee identified and evaluated the systems which are affected in a very minor way by operation of the plant at the uprated power level. These include the following systems:

- Extraction Steam
- Stator Cooling Water

Based upon NRC staff experience with previous reviews of power uprate applications and the fact that the extraction steam and stator cooling water systems do not perform any safety-related function, the NRC staff has not reviewed the impact of the proposed uprated power operations on the designs and performances of these systems.

3.14 Turbine-Generator

Evaluations for turbine operations with respect to design acceptance criteria to verify the mechanical integrity under the conditions imposed by the power uprate were performed. Results of the evaluations showed that there would be no increase in the probability of turbine overspeed nor associated turbine missile production due to plant operations at the proposed uprated power level. Therefore, the licensee concluded that the turbine could continue to be operated safely at the proposed uprated power levels.

Based on NRC staff review, the NRC staff finds that operation of the turbine at the proposed uprated power level is acceptable.

3.15 Miscellaneous Power Conversion Systems

The licensee had evaluated the miscellaneous steam and power conversion systems and their associated components (including the condenser air removal and steam jet air ejectors, turbine steam bypass, and feedwater and condensate systems) for plant operations at the proposed uprated power level. The licensee stated that the existing equipment for these systems are acceptable for plant operations at the proposed uprated power level.

Based upon previous NRC staff experience with previous power uprate reviews for similar BWR plants, the licensee's conclusion that existing equipment for these systems is acceptable for operation at the proposed uprated power level, and the fact that these systems do not perform any safety-related function, the NRC staff has not reviewed the impact of plant operations at the proposed uprated power level on the design and performance of these systems.

3.16 Liquid Waste Management

The liquid radwaste system is designed to process the majority of the liquid wastes within the plant so that liquids discharged from the plant satisfy the 10 CFR Part 20 and 10 CFR Part 50 Appendix I requirements. The activated corrosion products in liquid wastes are expected to increase proportionally to the power uprate. 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. The reduction does not affect plant safety. Reactor coolant cleanup flows, leaks, laboratory drains, dry solid waste, and spent resin quantities will remain essentially the same after power uprate. The licensee performed evaluations of plant operations and effluent reports, and concluded that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I will continue to be satisfied.

Based on NRC staff review of the licensee's rationale and the experience gained from review of power uprate applications for similar BWR plants, the NRC staff finds the liquid radwaste system acceptable.

3.17 Gaseous Waste Management

Gaseous wastes generated during normal and abnormal operation are collected, controlled, processed, stored, and disposed of utilizing the gaseous waste processing treatment systems. These systems which are designed to meet the requirements of 10 CFR Part 20 and 10 CFR Part 50 Appendix I include the offgas system and standby gas treatment system, as well as other building ventilation systems. Various devices and processes, such as radiation monitors, filters, isolation dampers, and fans, are used to control airborne radioactive gases. Results of licensee analyses demonstrate that airborne effluent activity released through building vents is not expected to increase significantly due to plant operations at the proposed uprated power level.

Based on NRC staff review of the licensee's rationale and the experience gained from review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the above systems.

3.18 Special Events

3.18.1 Station Blackout

Discussed under section 3.4.2.2 of this report.

3.18.2 High Energy Line Breaks (HELBs)

The slight increase in the reactor operating pressure and temperature resulting from the plant operations at the proposed uprated power level will cause a small increase in the mass and energy release rates following an HELB outside the primary containment. This results in a small increase in the subcompartment pressure and temperature profiles. The licensee conducted evaluations for the HELB in the piping systems (main steam, high pressure ECCS, RCIC, RWCU, CRD), and concluded that the existing HELB temperature and pressure analyses envelope those resulting from the proposed uprated power operation and that there is no change in postulated break locations due to plant operations at the proposed uprated power level.

Based on NRC staff review of the licensee's rationale and the experience gained from our review of power uprate applications for similar BWR plants, the NRC staff finds that the existing analysis for HELB remains bounding and is acceptable for plant operations at the proposed uprated power level.

3.18.3 Moderate Energy Line Break (MELB)

The licensee determined that uprated power level operation has no impact on the MELB. Based on a review of the high pressure ECCS, the RCIC system, the RWCU system, and the CRD system, the licensee concluded that the original MELB analysis is not affected by operation at the uprated power level.

Based on NRC review of the licensee's rationale and the experience gained from review of power uprate applications for similar BWR plants, the NRC finds that the existing analysis for MELB remains bounding and is acceptable for plant operations at the proposed uprated power level.

3.19 Equipment Qualifications (EQ)

The licensee evaluated the effects of plant operations at the proposed power level on qualified equipment including safety-related electrical equipment and mechanical components.

3.19.1 EQ of Electrical Equipment

3.19.1.1 Inside Containment

With regard to the temperature and pressure profiles used for qualifying equipment inside containment, the licensee stated that the results of existing calculations remain bounding for those temperatures and pressures resulting from plant operations at the proposed power level.

With regard to the radiation levels used for safety-related equipment qualification, the licensee stated that the existing calculated radiation levels were expected to increase 5 percent under normal plant operation and increase 8-10 percent under accident conditions. In Reference 9 the licensee reported that the qualification of any equipment which would be affected by higher radiation levels has been resolved. Previously, by Reference 6, the licensee had stated that qualification would be resolved by one or more of the following means: refined radiation calculations (location specific); slightly reducing qualified life; assessing the qualification based on actual test and material threshold data while maintaining the regulatory required margin, or assessing the impact of the radiation test and/or published threshold data on the material properties and its safety function.

Based on NRC staff review, the NRC staff finds the approach used by the licensee to resolve the qualification of safety-related electrical equipment inside containment for plant operations at the proposed uprated power level acceptable.

3.19.1.2 Outside Containment

With regard to the temperature and pressure profiles used for qualifying equipment outside containment, the licensee stated that the results of existing calculations remain bounding for the conditions resulting from plant operations at the proposed power level.

With regard to the radiation levels used for used for qualifying equipment outside containment, the licensee stated that the existing radiation levels were expected to increase 3% under normal plant operation and increase 5% under accident conditions. In Reference 9 the licensee reported that the qualification of any equipment which would be affected by higher radiation levels has been resolved. Previously, by Reference 6, the licensee had stated that qualification would be resolved by one or more of the following means: refined radiation calculations (location specific); slightly reducing qualified life; assessing the qualification based on actual test and material threshold data while maintaining the regulatory required margin, or assessing the impact of the radiation test and/or published threshold data on the material properties and its safety function.

Based on NRC staff review, the NRC staff finds the approach used by the licensee to resolve the qualification of safety-related electrical equipment outside containment for plant operations at the proposed uprated power level acceptable.

3.19.2 EQ of Mechanical Equipment with Non-Metallic Components

The licensee indicated that an evaluation of the non-metallic components of safety-related mechanical equipment shows that all equipment is bounded from the viewpoint of post-accident pressure, temperature, humidity, and dynamic loads. Some equipment was identified to be potentially impacted by the uprated radiation conditions.

By Reference 9 the licensee indicated that an evaluation of the non-metallic components of safety-related mechanical equipment is ongoing to show that all equipment is bounded from the standpoint of uprated radiation conditions. The licensee requested a license condition (license paragraph 2.L.(4) for Unit 1 and 2.I.(4) for Unit 2) as part of the power uprate amendment requiring that: environmental qualification of safety-related mechanical equipment with non-metallic components affected by uprated radiation conditions be resolved prior to the start-up of Unit 1 for Cycle 11 operation either by refined radiation calculations (location specific), slightly reducing qualified life, assessing the qualification based on actual test and material threshold data while maintaining the regulatory required margin, or assessing the impact of the radiation test and/or published threshold data on the material properties and its safety function.

Based on NRC staff review and the license condition, the staff finds the licensee's approach to resolve the qualification of the non-metallic components of safety-related mechanical equipment for plant operations at the proposed uprated power level acceptable.

3.20 Mechanical Component Review and Reactor Vessel Fracture Toughness

The staff's review of the safety analysis report provided by the licensee focused on the effects of power uprate on the structural and pressure boundary integrity of the piping systems and components, their supports, and reactor vessel and internal components and the control rod drive mechanism (CRDM), and the balance-of-plant (BOP) piping systems.

The GE generic guidelines (Ref. 9) for BWR power uprate were based on a 5 percent higher steam flow, an operating temperature increase of 5 °F, and an operating pressure increase of 40 psi or less. For BSEP, the maximum reactor vessel dome pressure increases from 1020 psia to 1045 psia (an increase of 25 psi), the dome temperature increases from 547 °F to 550 °F (an increase of 3 °F), and the steam flow rate increases from 10.470×10^6 lb_m/hr to 11.077×10^6 lb_m/hr for Units 1 and 2. The maximum core flow rate remains unchanged for the BSEP power uprate conditions.

3.20.1 Reactor Pressure Vessel and Internals

The licensee evaluated the RPV and internal components by considering load combinations that include reactor internal pressure difference (RIPD), loss-of-coolant-accident, and seismic loads. The seismic loads are unaffected by the power uprate. The licensee recalculated RIPDs for the power uprate shown in Tables 3-1, 3-2, and 3-3 of Reference 14, for normal, upset, and faulted conditions respectively.

The licensee evaluated the stresses and fatigue usage factors for reactor vessel components in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1965 Edition with addenda up to and including Summer 1967, for Units 1 and 2 to ensure compliance with the Code of record.

The maximum stresses for critical components were summarized in Table 3-4 of Reference 14. The load combinations for normal, upset, and faulted conditions were considered in the evaluation. The calculation was performed for the current shroud configuration, including repair brackets. Maximum stresses were calculated for the critical locations at the shroud repair bracket, the shroud, the shroud head assembly, and the shroud support. As shown in Table 3-4 (Ref. 14), the maximum stresses for the uprated power condition are less than the allowable stresses.

The cumulative usage factors (CUFs) for the uprated power level were calculated by using the power uprate scaling factor for limiting components such as feedwater nozzle, recirculation inlet nozzle, vessel shell and closure region bolts, and core spray nozzle. As provided in Table 3-5 of Reference 14, the maximum calculated fatigue CUFs are less than the allowable limit of 1.0.

With the power uprate, the CUF at the feedwater nozzle was changed from 0.75 to 0.86 for Unit 1 and from 0.73 to 0.96 for Unit 2. In References 5, 6, and

7, the licensee discussed in detail the differences in feedwater design and the analysis methods and assumptions for the two units. In addition, the licensee indicated that a fatigue analysis was performed at Brunswick based on the actual cycle transient data and projected 40-year feedwater nozzle CUF of 0.53 for Unit 1 and 0.21 for Unit 2. The CUF for the core spray nozzle for the power uprate was reduced from the current value of 0.98 to 0.96. In Reference 3, the licensee indicated that the power uprate CUF for the core spray nozzle was calculated by scaling up the maximum CUF for each region, and the current CUF of 0.98 was calculated based on the maximum enveloped stresses over all eight regions. The staff finds that the method used in calculating CUFs is acceptable.

On the basis of its review, the staff concludes that the maximum stresses and fatigue usage factors as provided by the licensee are within the allowable limits of the ASME Code and that the reactor vessel and internal components will continue to maintain structural integrity for the power uprate.

3.20.2 Control Rod Drive (CRD) System

The licensee evaluated the adequacy of the CRD mechanism in accordance with the Code of record, the ASME Code, Section III, 1968 Edition with Addenda up to and including Winter 1969 for Brunswick Unit 1, and 1968 Edition with addenda up to and including Winter 1968 for Brunswick Unit 2.

The licensee evaluated the CRD mechanism for the dome pressure of 1030 psig under the power uprate condition and an additional 40 psi for the vessel bottom head. The CRD mechanism was designed for a dome pressure of 1250 psig which bounds the uprated power condition.

As stated by the licensee, the maximum stress at the indicator tube of the CRD mechanism is 20,790 psi in comparison to the allowable stress of 26,060 psi. The maximum fatigue usage factor was calculated at the CRD main flange to be 0.15, which is far below the allowable limit of the ASME Code of 1.0.

On the basis of its review, the staff concludes that the CRD mechanism will continue to meet its design basis and performance requirements at uprated power conditions from the standpoints of stress and fatigue.

3.20.3 Reactor Coolant Piping and Components

The licensee evaluated the effects of the power uprate conditions, including higher flow rate, temperature and pressure for thermal expansion, fluid transients and vibration effects on the reactor coolant pressure boundary (RCPB), and the BOP piping systems and components.

The RCPB piping systems evaluated include main steam piping, reactor recirculation piping, the condensate and feedwater system, extraction steam lines, heater vents and drains, RWCU, RPV, head vent line, RCIC, HPCI, RHR, CS, and CRD piping. The components evaluated included equipment nozzles, anchors, guides, penetrations, pumps, valves, flange connections, and pipe supports.

The licensee evaluated the RCPB piping systems by reviewing the original design basis analysis against the increase of pressure, temperature, and flow for the power uprate. The percentage of increases in applicable code stresses, fatigue CUF, pipe interface component (including supports) loads, thermal and vibration displacements were calculated to determine scaling factors for the power uprate.

The maximum power uprate stresses and fatigue usage factors were computed by multiplying the scaling factors with the existing licensing basis values. The original code of record (USAS B31.1) as specified in the Brunswick UFSAR, the code allowables, and analytical techniques were used for the evaluation. No new assumptions were introduced that were not contained in the original analyses.

The licensee concluded that the existing licensing design basis analyses have sufficient margins between calculated stresses and Code limits to accommodate the increase in stresses resulting from higher operating flow, pressure, and temperature and, therefore, the power uprate will not have an adverse effect on the primary system piping and the system equipment design.

The licensee evaluated the stress levels for BOP piping and supports in a manner similar to the evaluation of the RCPB piping and supports based on increases in temperature and pressure of the design-basis analysis input. The evaluated BOP systems include lines which are affected by power uprate, but not evaluated in Section 3.5 of Reference 14, such as the main steam bypass lines, the main steam relief valve discharge, and portions of main steam and feedwater systems outside the primary containment. With the power uprate, the SRV loads, including SRV discharge line loads, for the power uprate increase approximately 0.1 percent, which is negligible in comparison to those used in the existing design-basis analysis defined as part of the Mark I containment long-term program for Brunswick.

The licensee concluded that there are sufficient margins between actual stresses and code limits in the existing licensing design basis and that all evaluated piping systems and components met the USAS B31.1 code stress criteria for the power uprate.

The licensee evaluated pipe supports, including anchorages, equipment nozzles, and penetrations, by comparing the increased piping interface loads on the system components resulting from the power uprate thermal expansion with the margin in the original design-basis calculation and, if necessary, performed detailed analyses using exact load combinations at the uprated conditions. The licensee also evaluated the effect of power uprate conditions on thermal and vibration displacement limits for struts, springs and pipe snubbers, and it was found to be acceptable.

The licensee reviewed the original postulated pipe-break analysis and concluded that the existing pipe-break locations were not affected by the power uprate, and no new pipe-break locations were identified.

On the basis of its review, the staff concludes that the design of piping, components and their supports will continue to meet the appropriate code criteria and will maintain the structural and pressure boundary integrity of the reactor coolant piping and supports in the power uprate conditions.

3.20.4 Equipment Seismic and Dynamic Qualification

On the basis of its review of the proposed power uprate amendment, the staff finds that the original seismic and dynamic qualification of the safety-related mechanical and electrical equipment is not affected by the power uprate conditions for the following reasons:

- Seismic loads are unchanged for the power uprate; and

- No new pipe-break locations resulted from the uprated conditions.

3.20.5 Conclusion for Mechanical Component Review

On the basis of its review, the staff finds that the licensee's proposed power uprate amendment has no adverse effect on the structural and pressure boundary integrity of piping systems, components, and their supports, reactor internals, core support structure, or the CRD system, and is, therefore, acceptable.

3.20.6 Reactor Pressure Vessel Fracture (RPV) Toughness

Operation with the 5% power uprate will result in a higher neutron flux at the reactor vessel wall, which would increase the integrated fluence over the period of plant life.

The licensee presented an assessment of the impact of power uprate on the RPV in Section 3.3.1 of Reference 14. In that assessment the licensee stated that BSEP's operating history results in a projected operation of 27 effective full power years (EFPY) at end of license. With a 5% power uprate, the projected operation is 28.5 EFPY. This is bounded by the UFSAR analysis basis of 32 EFPY; therefore, the licensee concluded that power uprate conditions are bounded by existing evaluations which show compliance with the current licensing basis and that the upper shelf energy (USE) will maintain the margin requirements of Appendix G to 10 CFR Part 50.

By telephone on October 25, 1996, the NRC staff requested that CP&L clarify (1) the potential impact of power uprate on the reactor material surveillance schedule, (2) whether power uprate impacts the existing pressure-temperature limit curves found in the TS, and (3) the potential impact of power uprate on compliance with 10 CFR Part 50, Appendices G and H with respect USE. The licensee provided the requested information in Reference 20.

Reactor Material Surveillance Schedule

10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements", requires a material surveillance program to monitor changes in the fracture toughness of RPV ferritic materials. Reactor licensees are required to meet 10 CFR 50, Appendix H unless otherwise approved by the NRC staff. 10 CFR 50, Appendix H requires that reactor beltline materials

surveillance programs must comply with ASTM E185-73, -79, or -82, as modified by Appendix H.

By letter dated July 6, 1992 (Serial: NLS-92-180), CP&L responded to NRC Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity." In response to an NRC question in the GL on actions being taken to comply with Appendix H of 10 CFR 50, CP&L stated that the ASTM E185-66 edition applied to the Brunswick Plant surveillance program conducted prior to removal of the first surveillance capsule. The CP&L response noted that a TS license amendment had been approved by the NRC staff to provide more useful neutron fluence data from the first surveillance capsule (TS Amendments 140 and 172 for Unit 1 and Unit 2, respectively), and that the removal schedule for the second and third surveillance capsules would be determined after removing the first capsules. In this respect, CP&L indicated that the revised schedule in the TS met the intent of ASTM E185-82.

In a subsequent response dated November 16, 1995 (Serial: BSEP 95-0572), to NRC GL 92-01, Revision 1, Supplement 1, CP&L provided initial and adjusted RT_{NDT} information based on NRC-approved methodology (which was approved by the NRC on December 16, 1994). Enclosure 3 to CP&L's November 16, 1995, letter also provided end-of-life (EOL) effective full power year (EFPY) values for both Brunswick Unit 1 and Unit 2 that were best estimates based on 24 month operating cycles, power uprate, and a thermal load factor of 97 percent. The estimated Unit 1 EOL EFPY was 30.5; the estimated Unit 2 EFPY was 29.3. As indicated in Reference 14, these values are bounded by the UFSAR analysis basis of 32 EFPY.

BSEP has existing RPV material surveillance processes that ensure the pressure-temperature P-T limit curves in the TS remain valid regardless of whether power uprate is or is not implemented. BSEP TS 3/4.4.6.1 provides the requirements for reactor coolant system pressure-temperature (P-T) limits as well as the surveillance schedule for reactor material irradiation specimens.

BSEP TS 4.4.6.1.3 requires that the cumulative EFPY be determined at least once every 18 months. BSEP TS 4.4.6.1.3 also requires that reactor material irradiation specimens be removed and examined in accordance with the schedule specified in TS Table 4.4.6.1.3-1. To date, one Unit 1 specimen has been removed, examined, and the results documented in a report submitted to the NRC by letter dated August 17, 1994 (Serial: BSEP 94-0316). In addition, one Unit 2 specimen has been removed (during the February 1996 refueling outage) and its evaluation is in progress.

The Unit 1 surveillance report describes the "remaining surveillance program" by stating that Unit 1 has two capsules remaining, that removal dates for the two remaining capsules have not yet been selected because of the desire to establish an integrated surveillance program for both Unit 1 and Unit 2, and that it is desirable to remove and examine the first Unit 2 surveillance specimen prior to proposing an integrated surveillance program. TS Table 4.4.6.1.3-1, note (b), requires that the withdrawal schedules for the second and third specimens will be established following examination of the first specimen.

Pressure-Temperature Limit Curves

10 CFR Part 50, Appendix H, Section IV (Report of Test Results) specifies that the results of surveillance capsule tests must be submitted to the NRC within one year of capsule withdrawal and that a schedule must be provided for the submittal of changes to the TS P-T curves, if required. Thus, based on this requirement, CP&L must submit the Unit 2 surveillance capsule test results within the required time frame along with a schedule for the submittal of any change(s) that may be needed to the TS P-T curves.

The existing Unit 1 and Unit 2 P-T curves for normal operations (shown in TS Figures 3.4.6.1-2 and 3.4.6.1-1) are based on Unit 2 reactor cavity neutron fluence dosimetry obtained prior to removal of the first reactor material specimen. For Unit 1, the fluence from the reactor cavity dosimetry has been shown to be approximately 34 percent higher than the fluence data obtained from the reactor material specimen. For Unit 2, preliminary data shows that the fluence from the reactor cavity dosimetry is approximately 24 percent higher than the fluence data obtained from the reactor material specimen. Therefore, even accounting for the slightly higher neutron fluence that may result from power uprate, the existing P-T curves are expected to remain conservative and valid through 16 EFPY and are applicable to operation both at the currently licensed power level and at the proposed power uprate level. As of September 1996, Unit 1 had accumulated approximately 10.95 EFPY, and Unit 2 had accumulated approximately 11.40 EFPY. Performance of the 18-month periodic review to determine the cumulative EFPY ensures that CP&L is aware of the current status of reactor vessel exposure relative to the 16 EFPY limit specified on the operating P-T curves in the TS.

Upper Shelf Energy

10 CFR Part 50, Appendix G, Section IV, Item A.1.a states that reactor beltline materials must have Charpy USE in the transverse direction for the base material and along the weld for weld material of no less than 75 ft-lb initially and must maintain Charpy USE throughout the life of the vessel of no less than 50 ft-lb. This section also states that lower values of Charpy USE are acceptable if demonstrated to provide margins of safety against fracture equivalent to those required by the ASME Code, Appendix G.

The BWR Owners' Group (BWROG) has previously submitted and obtained NRC approval of a topical report entitled "10 CFR 50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy In BWR/2 Through BWR/6 Vessels." This program is applicable to plate and weld material. By letter dated May 13, 1994 (Serial: BSEP 94-0179), CP&L documented the applicability of this methodology to the BSEP reactor vessels. In applying the BWROG methodology, CP&L used estimated EOL fluence values that were based on preliminary Unit 1 surveillance specimen data. These fluence values bound the most recent EOL fluence projections provided in CP&L's letter responding to NRC Generic Letter 92-01, Revision 1, Supplement 1 dated November 16, 1995 (Serial: BSEP 95-0572), which include consideration of both power uprate and 24-month fuel cycles.

In the November 16, 1995 letter, CP&L indicated that the N16 nozzles were not expected to drop below 50 ft-lb due to the low expected EOL fluence. As discussed in CP&L's November 16, 1995 letter, a plant-specific USE equivalent margins analysis for these nozzles is being completed which will include estimated projections of EOL USE. The fluence projections considered in this analysis bound EOL, including 24-month fuel cycles and power uprate. Preliminary data from this analysis confirms that EOL USE on these nozzles will exceed the 50 ft-lb. minimum criteria specified in 10 CFR 50, Appendix G, Section IV. Additionally, the preliminary analysis results also indicate that equivalent margins of safety can be demonstrated for these nozzles with USE values as low as 30 ft-lb.

Based upon (1) the BSEP TS requirement to establish the schedule for withdrawl of the remaining surveillance capsules following completion of examination of the first specimen (an examination which is completed on Unit 1 and ongoing on Unit 2), (2) the 10 CFR Part 50 Appendix H requirement for a surveillance program meeting ASTM E185-73, 79, or 82, (3) the 10 CFR 50 Appendix H requirement to provide a schedule for any needed changes to the P-T curves within a year of specimen analysis, (4) the expectancy that the current P-T curves will remain valid to 16 EFPY under uprated conditions and the units are only at 10.95 (Unit 1) EFPY and 11.4 (Unit 2) EFPY, and (5) the satisfactory EOL USE values, the NRC staff finds that the affects of power uprate on RPV fracture toughness have been adequately considered.

3.21 Instrumentation and Control

The licensee's submittals provided information regarding the methodology used for setpoint calculations and a plant-specific methodology contained in Reference 14 and its supplement (Ref. 15). This report uses the generic format and methodology documented in References 11 and 13. The staff has previously reviewed and accepted References 11 and 13.

The proposed setpoint changes resulting from the power uprate are intended to maintain the existing margins between operating conditions and reactor trip setpoints and do not significantly increase the likelihood of a false trip or failure to trip upon demand. Therefore, the existing licensing basis is not affected by the setpoint changes to accommodate the power uprate.

3.21.1 Technical Specifications (TS) Changes

The following TS changes related to power uprate have been proposed by the licensee:

Change 1

1. Rated Thermal Power (Facility Operating License DPR-71 section C.1, page 3 and TS Definitions, page 1-6)
   Change RATED THERMAL POWER from 2436 MWt to 2558 MWt.

The increase of rated thermal power from 2436 MWt to 2558 MWt follows the generic guidelines of Reference 11, which provide the generic licensing criteria, methodology, and scope of evaluations and reviews to be performed to demonstrate the ability to operate safely at an uprated power level.

Changes 2, 3, and 23

2. Flow Biased Simulated Thermal Power - High (Table 2.2.1-1.2b, page 2-4 and Figure 2.2.1-1, page 2-6)
Change Trip Setpoint from ≤(0.66W + 64%) with a maximum ≤113.5% of RATED THERMAL POWER to ≤(0.66W + 59.6%) with a maximum ≤113.6% of RATED THERMAL POWER.
Change Allowable Value from ≤(0.66W + 67%) with a maximum ≤115.5% of RATED THERMAL POWER to ≤(0.66W + 61%) with a maximum ≤115.3% of RATED THERMAL POWER.
Change the power/flow boundary of Figure 2.2.1-1.

3. Fixed Neutron Flux - High (Table 2.2.1-1.2c, page 2-4)
Change Trip Setpoint from ≤120% of RATED THERMAL POWER to ≤116.3% of RATED THERMAL POWER.
Change Allowable Value from ≤120% of RATED THERMAL POWER to ≤118% of RATED THERMAL POWER.

23. Average Power Range Monitor (APRM) Upscale (Flow Biased) Rod Block (Table 3.3.4-2.1.a, page 3/4 3-50)
Change Trip Setpoint from ≤(0.66W+58%) with a maximum of ≤108% of RATED THERMAL POWER to ≤(0.66W+54.6%) with a maximum of ≤109.3% of RATED THERMAL POWER.
Change Allowable Value from ≤(0.66W+61%) with a maximum of ≤110% of RATED THERMAL POWER to ≤(0.66W+56%) with a maximum of ≤111% of RATED THERMAL POWER.

The analytical limits for the Flow Biased Simulated Thermal Power - High, Fixed Neutron Flux - High scram functions and APRM Upscale Rod Block are not affected by power uprate since they are based on a percentage of rated thermal power. The TS Allowable Value and Trip Setpoint have been recalculated due to the change in the power/flow map boundary. The power/flow map boundary changes are based on the change from the existing 100 percent thermal power to 105 percent thermal power. The boundary is clamped so that any increase in flow will not result in an increase in thermal power. The flow rate at which this clamp occurs changes from 75 percent to 81 percent under power uprate. The revised TS Allowable Value and Trip Setpoint ensure adequate margin exists to the analytical limit.

Change 4

4. Reactor Vessel Steam Dome Pressure - High (Table 2.2.1-1.3, page 2-4)
Change Trip Setpoint from ≤1045 psig to ≤1067.9 psig.
Change Allowable Value from ≤1045 psig to ≤1070 psig.

Satisfactory reactor vessel pressure control requires an adequate flow margin between the uprated operating conditions and the steam flow capability of the turbine control valves at their maximum stroke. Therefore, the reactor vessel steam dome operating pressure for uprated power is being increased by 25 psi to assure that satisfactory reactor pressure control is maintained. Since the reactor vessel steam dome operating pressure is being increased, the reactor vessel steam dome high pressure scram is also being increased the same amount to preserve the existing margins to reactor trips.

Changes 5, 6, and 9

5. Reactor Vessel Water Level - Low Level 1 (Tables 2.2.1-1.4, page 2-4, 3.3.2-2.1.a.1, page 3/4 3-18, 3.3.2-2.5.a, page 3/4 3-22, and 3.3.3-2.4.c, page 3/4 3-40)
Change Trip Setpoint from ≥+162.5 inches to ≥+153.2 inches.
Change Allowable Value from ≥+162.5 inches to ≥+153 inches.

6. Reactor Vessel Water Level - Low, Level 3 (Tables 3.3.2-2.1.a.2, page 3/4 3-18, 3.3.3-2.1.a, page 3/4 3-39, 3.3.3-2.2.b, page 3/4 3-39, and 3.3.3-2.4.b, page 3/4 3-40)
Change Trip Setpoint from ≥+2.5 inches to ≥+14.1 inches.
Change Allowable Value from ≥+2.5 inches to ≥+13 inches.

9. Reactor Vessel Water Level - Low, Level 2 (Tables 3.3.2-2.2.c, page 3/4 3-19, 3.3.2-2.3.e, page 3/4 3-19, 3.3.3-2.3.a, page 3/4 3-40, 3.3.6.1-2.1, page 3/4 3-90, and 3.3.7-2.1, page 3/4 3-95)
Change Trip Setpoint from ≥+112 inches to ≥+104.1 inches.
Change Allowable Value from ≥+112 inches to ≥+103 inches.

Power uprate normal operating pressures and temperatures were used in CP&L's setpoint methodology to calculate the Allowable Values and Trip Setpoints for the reactor vessel water level instruments. The revised Allowable Values and Trip Setpoints are consistent with the analytical limits determined by GE Topical Report NEDC-31624P, "Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," Revision 2 and OPL-3, "Plant Operating Licensing Analysis for Brunswick Steam Electric Plant Unit 1."

Changes 7, 10, 13, and 31

7. Main Steam Line Flow - High (Table 3.3.2-2.1.c.3, page 3/4 3-18)
Change Trip Setpoint from ≤140% of rated flow to ≤137% of rated flow.
Change Allowable Value from ≤140% of rated flow to ≤138% of rated flow.

10. HPCI Steam Line Flow - High (Table 3.3.2-2.4.a.1, page 3/4 3-20)
Change Trip Setpoint from ≤300% of rated flow to ≤272% of rated flow.
Change Allowable Value from ≤300% of rated flow to ≤275% of rated flow.

13. RCIC Steam Line Flow - High (Table 3.3.2-2.4.b.1, page 3/4 3-21)
Change Trip Setpoint from ≤300% of rated flow to ≤272% of rated flow.
Change Allowable Value from ≤300% of rated flow to ≤275% of rated flow.

31. Main Steam Line Flow - High (Table 3.3.2-2.1.c.4, page 3/4 3-18) Unit 2 only
Change Trip Setpoint from ≤40% of rated flow to ≤30% of rated flow.
Change Allowable Value from ≤40% of rated flow to ≤32% of rated flow.

Changes for these functions are a direct result of power uprate. The Analytical Limits for these functions do not change as a result of power uprate, since they are based on a percentage of flow. CP&L's setpoint methodology and the uprated design flow were used to determine the revised TS Allowable Values and Trip Setpoints.

Change 24

24. Reactor Vessel Pressure - High (Table 3.3.6.1-2.2, page 3/4 3-90)
Change Trip Setpoint from ≤1120 psig to ≤1137.8 psig.
Change Allowable Value from ≤1120 psig to ≤1143 psig.

The Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) for the high reactor vessel pressure trip function initiates a trip of the recirculation pump in the event of an ATWS transient. The ATWS-RPT related reactor vessel high pressure Allowable Value and Trip Setpoint are being increased to account for the increase in reactor vessel operating pressure and safety relief valve setpoints caused by power uprate and to maintain operating margin. Raising the ATWS-RPT related reactor vessel high pressure Allowable Value and Trip Setpoint prevents unnecessary recirculation pump trips following pressurization transients with a reactor scram while maintaining the requirements of 10 CFR 50.62.

Change 25

25. Reactor Vessel Water Level - High (Table 3.3.7-2.2, page 3/4 3-95)
Change Trip Setpoint from ≤+208 inches to ≤+206.8 inches.
Change Allowable Value from ≤+208 inches to ≤+207 inches.

The Analytical Limit for this function does not change as a result of power uprate. The Allowable Value and Trip Setpoint has been recalculated in accordance with CP&L's setpoint methodology.

Change 26

26. Thermal Power Limitations (Figure 3.4.1.1-1, page 3/4 4-1b)
Rescale Figure 3.4.1.1-1 by 1/1.05 based on the change in reactor thermal power.

The change in reactor thermal power necessitates a change in the scale for this figure. With power uprate there is no increase in the core flow; therefore, the figure is shifted downward by the ratio of current rated thermal power to uprate thermal power (1/1.05).

Change 28

28. Reactor Steam Dome Pressure (Specifications 3.4.6.2, page 3/4 4-21, and 4.4.6.2, page 3/4 4-21, and Bases 3/4.6.2, page B 3/4 6-3)
Change Allowable Value from <1020 psig to <1045 psig.

The maximum reactor vessel dome pressure is an initial condition of the reactor vessel over pressure protection analysis. For power uprate, the over pressure analysis assumed an increase in initial reactor vessel dome pressure of 25 psi. Therefore, the new reactor vessel dome pressure Allowable Value is <1045 psig.

Changes 29 and 30

29. HPCI Operability Test (Specification 4.5.1.b, page 3/4 5-2)
Change Allowable Value from reactor pressure ≥1000 psig to reactor pressure ≥1025 psig and from turbine at 1000 psig +20, -80 psig to turbine at 1025 psig +20, -80 psig.

30. RCIC Operability Test (Specification 4.7.4.b, page 3/4 7-7)
Change Allowable Value from turbine at 1000 psig +20, -80 psig to turbine at 1025 psig +20, -80 psig.

The allowable HPCI and RCIC surveillance test pressure is being increased by 25 psi to correspond with the increase in normal reactor operating pressure. This change will allow the quarterly demonstration of HPCI and RCIC capability to be performed at normal reactor operating pressures, which meets the intent of the current TS.

The following TS changes based on reconstituted calculations, using revised calibration frequency, plant calibration tolerances, and power uprate, have been proposed by the licensee:

Changes 8, 11, 12, and 14 - 22

8. Condenser Vacuum - Low (Table 3.3.2-2.1.e, page 3/4 3-18)
Change Trip Setpoint from ≥7 inches Hg vacuum to ≥7.6 inches Hg vacuum.
Change Allowable Value from ≥7 inches Hg vacuum to ≥7.5 inches Hg vacuum.

11. HPCI Steam Supply Pressure - Low (Table 3.3.2-2.4.a.3, page 3/4 3-20)
Change Trip Setpoint from ≥100 psig to 106.6 psig.
Change Allowable Value from ≥100 psig to ≥104 psig.

12. HPCI Turbine Exhaust Diaphragm Pressure - High (Table 3.3.2-2.4.a.6, page 3/4 3-20)
Change Trip Setpoint from ≤10 psig to ≤8.5 psig.
Change Allowable Value from ≤10 psig to ≤9.0 psig.

14. RCIC Steam Supply Pressure - Low (Table 3.3.2-2.4.b.3, page 3/4 3-21)

Change Trip Setpoint from ≥50 psig to ≥55.6 psig.
Change Allowable Value from ≥50 psig to ≥53 psig.

15. RCIC Turbine Exhaust Diaphragm Pressure - High (Table 3.3.2-2.4.b.6, page 3/4 3-21)
Change Trip Setpoint from ≤10 psig to ≤5.5 psig.
Change Allowable Value from ≤10 psig to ≤6.0 psig.

16. Reactor Steam Dome Pressure - High (Table 3.3.2-2.5.b, page 3/4 3-22)
Change Trip Setpoint from ≤140 psig to ≤130.8 psig.
Change Allowable Value from ≤140 psig to ≤137 psig.

17. Reactor Steam Dome Pressure - Low (Table 3.3.3-2.1.b, page 3/4 3-39)
Change Trip Setpoint from 410 ±15 psig to ≥406.7 psig.
Change Allowable Value from 410 ±15 psig to ≥404 psig.

18. Reactor Steam Dome Pressure - Low/RHR Pump Start and LPCI Valve Actuation (Table 3.3.3-2.2.d.1, page 3/43-39)
Change Trip Setpoint from 410 ±15 psig to ≥406.7 psig.
Change Allowable Value from 410 ±15 psig to ≥404 psig.

19. Reactor Steam Dome Pressure - Low/Recirculation Pump Discharge Valve Actuation (Table 3.3.3-2.2.d.2, page 3/4 3-39)
Change Trip Setpoint from 310 ±15 psig to ≥306.7 psig.
Change Allowable Value from 310 ±15 psig to ≥304 psig.

20. ADS Timer (Table 3.3.3-2.4.d, page 3/4 3-40)
Change Trip Setpoint from ≤120 seconds to ≤83 seconds.
Change Allowable Value from ≤120 seconds to ≤108 seconds.

21. Core Spray Pump Discharge Pressure - High (Table 3.3.3-2.4.e, page 3/4 3-40)
Change Trip Setpoint from ≥100 psig to ≥112.1 psig.
Change Allowable Value from ≥100 psig to ≥102 psig.

22. RHR (LPCI Mode) Pump Discharge Pressure - High (Table 3.3.3-2.4.f, page 3/4 3-40)
Change Trip Setpoint from ≥100 psig to ≥111.1 psig.
Change Allowable Value from ≥100 psig to ≥102 psig.

Changes for the above functions are not a direct result of power uprate. These changes are included to eliminate the need for additional administrative control due to reduced margin and the increased potential for inadvertent safety function actuation that would be caused by the implementation of power uprate without these changes. The calculations were performed using CP&L's setpoint methodology to determine the new Allowable Values and Trip Setpoints. The changes in Allowable Values for these functions are in a more conservative direction, providing additional margin.

Changes 17, 18, and 19 have been made to replace the range currently shown in the TS with the minimum required pressure for these functions. These functions are permissives, which ensure that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure.

3.21.2 Conclusion Regarding Instrumentation and Control Review

Based on the above evaluation and the fact that the proposed setpoint changes resulting from the power uprate are intended to maintain the existing margins between operating conditions and reactor trip setpoints and do not significantly increase the likelihood of a false trip or failure to trip upon demand, the staff concludes that the licensee's proposed instrument setpoint changes incorporated in the TS for power uprate are consistent with the Brunswick setpoint methodology, licensing basis, and BWR Standard Technical Specifications, and are, therefore, acceptable.

3.22 Radiological Effects of Power Uprate

The licensee evaluated the impact of the proposed amendment to show that the applicable regulatory acceptance criteria continue to be satisfied for the uprated power conditions. In conducting this evaluation, the licensee re-evaluated the effect of the power uprate on DBA radiological consequences. The original licensing DBA source terms for Brunswick were considered. The licensee also re-evaluated the control room habitability under DBA conditions.

The licensee stated that the radiological consequence analyses were performed using standard models developed by General Electric (GE) that have been utilized in other power uprate projects. The dose analyses were based on plant-specific parameters from the UFSAR and were done at both the current power and at 102 percent of the proposed uprate power. The licensee's analyses indicate that the calculated offsite radiological consequences doses for all DBAs are within the dose acceptance criteria stated in the Standard Review Plan (SRP) (NUREG 0800) and 10 CFR Part 100 and also comply with the dose acceptance criteria for control room operators given in GDC 19 of Appendix A to 10 CFR Part 50.

The staff performed confirmatory evaluations of radiological consequences of DBAs for the proposed power uprate. To account for the power uprate, the results from the original Brunswick Safety Evaluation Report (SER) from November 1973 were increased by 5 percent. A comparison of the results shows the licensee's results are significantly lower than staff calculations. When examined, this difference can be ascribed to the different methods used by the licensee and staff for calculating the atmospheric dispersion factors (X/Q) used in the dose calculations. The staff's analysis used X/Q values from the original Brunswick SER which are a part of the licensing basis. The staff evaluated the large LOCA, control rod drop accident (CRDA), and the fuel handling accident (FHA) for the power uprate. The whole body and thyroid doses were calculated for the exclusion area boundary and the low population zone. The main steam line break outside of containment (MSLB) accident was determined to be limiting for control room dose in an SER dated February 16, 1989. The doses from this subsequent SER were increased by 5 percent to account for the power uprate. The staff control room doses are shown compared to the licensee's results due to a LOCA. The results for the DBAs are tabulated below.

Uprate DBA Radiological Consequences			
LOCA	Dose (rem)		
	CPL @ 2609 MWt	NRC @ 2678 MWt	Acceptance Criteria
Exclusion Area Boundary			
Whole Body	0.29	3.2	25
Thyroid	1.07	42.0	300
Low Population Zone			
Whole Body	0.32	2.1	25
Thyroid	4.02	41.0	300
Control Room			
Whole Body	0.13	< 0.1 (1)	5
Thyroid	3.26	20.0 (1)	30

(1) Control room doses for MSLB.

Uprate DBA Radiological Consequences			
CRDA	Dose (rem)		
	CPL @ 2609 MWt	NRC @ 2678 MWt	Acceptance Criteria
Exclusion Area Boundary			
Whole Body	0.097	1.1	6
Thyroid	1.36	21.0	75
Low Population Zone			
Whole Body	0.041	1.1	6
Thyroid	1.17	23.1	75

Uprate DBA Radiological Consequences			
FHA	Dose (rem)		
	CPL @ 2609 MWt	NRC @ 2678 MWt	Acceptance Criteria
Exclusion Area Boundary			
Whole Body	0.034	< 1.1	6
Thyroid	0.036	2.1	75
Low Population Zone			
Whole Body	0.015	< 1.1	6
Thyroid	0.016	1.1	75

The staff finds that the offsite radiological consequences and control room operator doses for all DBAs at the uprated power level of 2558 MWt will continue to meet the acceptance criteria of the SRP, 10 CFR Part 100, and GDC 19. Therefore the staff, with regard to this aspect of its review, concludes that the licensee's request to uprate the authorized maximum core power level by 5 percent to 2558 MWt from its current limit of 2436 MWt is acceptable.

3.23 Human Factors

The staff reviewed Carolina Power and Light Company's (CP&L's) submittals (Refs. 1-10) for power uprate. The staff's evaluation of the licensee's responses relative to five review topics is provided below.

Topic 1 - Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. By Reference 2, the licensee stated that changes to emergency and abnormal operating procedures required for the power uprate will only include revision to previously defined numerical values (e.g., RPV high pressure scram setpoint value). Further, the licensee stated that the power uprate would not change the type, scope, and nature of operator actions needed for accident mitigation and that it would not require any new operator actions. The staff finds that the licensee's responses are satisfactory.

Topic 2 - Provide examples of operator actions potentially sensitive to power uprate and address whether the power uprate will have any effect on operator reliability or performance. By Reference 5, the licensee identified the following operator actions that would necessitate reduced response times associated with a power uprate: (1) manually depressurizing the reactor following a loss of high pressure injection, and (2) injecting boron through the Standby Liquid Control (SLC) system during an ATWS. The licensee stated that operators must manually depressurize the reactor within about 30 minutes and that the reduction in available operator response time would be approximately 90 seconds. The licensee also stated that operators must initiate SLC within about 10 minutes and that the reduction in available operator response time would be about 30 seconds. The licensee noted that simulator observations have shown that operators will typically actuate the

SLC system in two to three minutes in response to the ATWS event and manually depressurize the reactor if there is a loss of all high pressure injection in less than a third of the available time. By letter Reference 6 the licensee clarified that the reduced operator response times are needed because the plant parameters being monitored by the operator, that would indicate that the previously discussed manual actions (i.e., depressurize and initiate the SLC system) are needed, would be reached sooner due to the reactor operating at a higher power; therefore, operator action would also be needed sooner. On the basis of previously discussed information, the staff finds that the reduction in time available to the operator due to the power uprate is relatively small and it should not significantly affect the operator's ability to complete the subject manual actions in the times required.

Topic 3 - Discuss any changes the power uprate will have on control room instruments, alarms, and displays. Are zone markings on meters changed (e.g., normal range, marginal range, and out-of-tolerance range)? By Reference 5, the licensee stated that there are six control room indicators that will require a face-plate-scale change and two indicators that will require a change to normal-range zone markings. The licensee noted that there are no changes needed for the marginal and out-of-tolerance ranges. The licensee indicated that various instructional aids (e.g., labels, sketches, and markings) in the main control room will be revised due to the power uprate. The licensee stated that the previously discussed changes would be communicated to operators through training. The staff finds that the licensee's responses are satisfactory.

Topic 4 - Discuss any changes the power uprate will have on the Safety Parameter Display System (SPDS). By letter Reference 5, the licensee stated that the changes to the SPDS would include the following: input and output points, constants which feed composed points, SRV lift setpoints, and Emergency Operating Procedure (EOP) limit graphs. Further, the licensee stated that these changes would be transparent to the operator and EOP execution will not be affected. The staff finds that the licensee's responses are satisfactory.

Topic 5 - Describe any changes the power uprate will have on the operator training program and the plant simulator. By Reference 9, the licensee requested the following license condition (license paragraph 2.L.(5) for Unit 1 and 2.I.(5) for Unit 2) associated with power uprate:

1. Classroom Training

   Power Uprate Operator Training, including the plant operating parameter changes resulting from power uprate, shall be performed as part of License Operator Retraining (LOR) prior to Unit 1 start-up for Cycle 11 operation.

2. Simulator Training

   Simulator training for power uprate shall be completed prior to Unit 1 start-up for Cycle 11 operation. The simulator training will include the following:

   (i) A demonstration of selected transients at the uprated power compared to the non-uprated power, including changes in time to achieve critical points for operator actions.

   (ii) The time to meet the conditions to inject boron for a high power ATWS.

   (iii) The time to depressurize the reactor on a loss of all high pressure injection (time to achieve conditions requiring emergency depressurization at TAF; non-ATWS).

3. Simulator Modification

   Prior to Unit 1 start-up for Cycle 11, the simulator shall be modified to match the uprated control room, as close as possible, with the exception of the zone coding for the HPCI System and RCIC System speed indication meters. The HPCI and RCIC speed indication zone coding shall be completed prior to Unit 2 start-up following the implementation of the power uprate license amendment.

On the basis of the information discussed and the license condition, the staff finds that the licensee has proposed satisfactory changes to the operator training program and the plant simulator related to the power uprate.

The staff concludes that the previously discussed review topics associated with the proposed Brunswick Steam Electric Plant Units 1 and 2 power uprate have been satisfactorily addressed. The staff further concludes that the power uprate should not adversely affect operator performance or operator reliability.

3.24 Startup Testing

In Reference 14 the licensee stated that a startup testing program will be performed which is consistent with the NRC-approved guidelines described in Reference 13. Because the licensee's approach conforms to the NRC-approved guidelines, the NRC staff finds the startup testing program acceptable.

3.25 Evaluation of Effect on Responses to Generic Communications, Plant-Unique Safety Evaluations, and Temporary Modifications

In Reference 13 GE submitted a generic assessment of the effect of power uprate on responses to generic NRC and industry communications. The NRC staff approved the assessment in a letter dated July 31, 1992 (Reference 19). Additionally the licensee and GE performed a similar review of the effects of power uprate on plant-specific responses to NRC and Industry communications and performed an evaluation of the effects of power uprate on safety evaluations for work in progress and on temporary modifications. Any items found unacceptable were revised. The staff may audit these activities after plant startup following the implementation of power uprate. The staff finds this approach consistent with the NRC-approved generic guidelines in Reference 11 and therefore acceptable.

3.26 Flow-Accelerated Corrosion (FAC) in Piping

Erosion/corrosion (also referred to as flow accelerated corrosion or FAC) in piping may be affected by the increased flow rates, higher operating temperatures, and change in moisture content of two-phase flow streams. In Reference 14 the licensee presented the results of its evaluation of the impact of power uprate on piping FAC. The licensee stated that the differences in operating conditions due to power uprate are small and should not cause a new FAC concern. BSEP has a formal FAC program for monitoring pipe wall thickness, and the licensee stated that any increase in the rate of pipe wall thinning that occurs as a result of the uprate will be identified through this program.

The licensee uses industry experience, plant experience, and the Electric Power Research Institute (EPRI) CHECKMATE model as the primary bases for the BSEP FAC program. CP&L used the CHECKMATE model to rank systems and components with respect to FAC susceptibility. By telephone on October 25, 1996, the NRC staff requested that CP&L address whether BSEP will experience a higher steam moisture content as a result of power uprate and whether the higher flow (and, if applicable, higher moisture content) could change the FAC monitoring/susceptibility points as determined by the CHECKMATE model. The licensee responded by Reference 20, and stated that the expected increase in steam moisture content as a result of power uprate is considered insignificant (i.e., $\leq$ 0.03 percent), with the overall steam quality from the reactor pressure vessel decreasing from 99.59% to 99.56%. A 0.03 percent increase in steam moisture content would have less than a 0.5 percent change in the wear-rate calculations. This small change would not result in a change to the monitoring points or affect the FAC model. The approximately 6 percent increase in steam line flow resulting from power uprate will not have a significant effect on FAC. The expected flow increase will have a negligible effect on wear rates (i.e., 1 to 2 percent). Therefore, CP&L does not expect power uprate to have an impact on the BSEP FAC program.

Because the operating conditions under power uprate will not affect the monitoring points for FAC, the expected increases in pipe wall wear rates are small, and the FAC program will continue to monitor for increases in pipe wall wear rates, the NRC staff finds that the effects of power uprate on FAC have been adequately considered and are not significant.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on October 28, 1996 (61 FR 55673). Accordingly, based on the Environmental Assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, "105% Thermal Power Uprate," April 2, 1996 (BSEP 96-0123).

2. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, "Power Uprate," November 20, 1995 (BSEP 95-0535).

3. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, July 1, 1996 (BSEP 96-0242).

4. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, July 30, 1996 (BSEP 96-0287).

5. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, August 7, 1996 (BSEP 96-0300).

6. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, September 13, 1996 (BSEP 96-0340).

7. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, September 20, 1996 (BSEP 96-0348).

8. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, October 1, 1996 (BSEP 96-0362).

9. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, October 22, 1996 (BSEP 96-0392).

10. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, October 22, 1996 (BSEP 96-0403).

11. 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.

12. W. T. Russell, U.S. Nuclear Regulatory Commission, letter to P.W. Marriott, General Electric Company, "Staff Position Concerning General Electric Boiling Water Reactor Power Uprate Program," September 30, 1991.

13. 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 and 2.

14. GE Nuclear Energy, "Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant, Units 1 & 2," Licensing Topical Report NEDC-32466P, Class III (Proprietary), September 1995; NEDO-32466, Class I (Non-proprietary), September 1995.

15. GE Nuclear Energy, "Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant, Units 1 & 2 (Supplement 1)," Licensing Topical Report NEDC-32466P, Class III (Proprietary), March 1996; NEDO-32466, Class I (Non-proprietary), March 1996.

16. "Brunswick Steam Electric Plant, Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (Revision 2)," NEDC-31624P, July 1990.

17. "Brunswick Steam Electric Plant, Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis: Application to GE 13 Fuel," NEDC-31624P, Supplement 3, Revision 0, January 1996.

18. NUREG-0661, "Mark I Containment Long Term Program Safety Evaluation," July 1980.

19. W. T. Russell, U.S. Nuclear Regulatory Commission, letter to P.W. Marriott, General Electric Company, "Staff Safety Evaluation of General Electric Boiling Water Reactor Power Uprate Generic Analysis," July 31, 1992.

20. W. R. Campbell, Carolina Power & Light Company, letter to U.S Nuclear Regulatory Commission, October 29, 1996 (BSEP 96-0412).

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Date: November 1, 1996