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

ATTACHMENT

SAFETY EVALUATION REPORT
OFFICE OF NUCLEAR REACTOR REGULATION
EXTENDED POWER OPERATION
SOUTHERN NUCLEAR CORPORATION
HATCH UNITS 1 AND 2

I. INTRODUCTION

Southern Nuclear Operating Company, Inc. (SNC), the licensee for Edwin 1. Hatch, Units 1 and 2, submitted a request by letter dated August 8, 1997 to increase the licensed thermal power level from current 2558 MWt to 2763 MWt. SNC had previously requested 5 % power uprate for both units, and NRC approved the power uprate from the original licensing power rate of 2436 MWt to 2558 MWt in a letter dated August 31, 1995. The proposed power uprate represents a 8% increase from the current power and a 13% power increase from the original licensed power of 2436 MWt. In addition to the August 1997 submittal, SNC also provided additional information in a letter dated March 9, 1998. in response to a request for additional information from the staff.

The proposed power uprate is accomplished by extending the power-flow map within (approximately) equivalent rod and core flow control lines. The proposed rod and flow control line for the 13% power increase corresponds to the 120% rod and flow control line relative to the original licensed power of 2436 MWt and approximately 115% rod and flow control line relative to the current 5% increase power of 2558 MWt. The licensee also stated that the proposed rod and flow control line is consistent to the MELLA rod and flow control line of the original licensed power and that with the extended power uprate, the highest analyzed rod and flow control line will be no higher than that of a BWR/4 MELLA plant at the original licensed power.

The planned proach 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) no increase in reactor operating pressure relative to the 5 % power uprate, and (5) reactor operation primarily along equivalent rod and flow control lines.

SNC proposed to achieve the power uprate by supplying the higher steam flow to the turbine generator. The licensee had modified the high pressure turbine tc accommodate the higher steam supply. SNC also states that: (1) due to improvements in the analytical techniques (i.e. computer codes and data) for several decades of BWR safety technology; (2) plant performance feedback; and (3) improved fuel and core design have resulted in an significant increase in the margin between the calculated safety analysis and the licensing limits. Thus, increase in the safety analysis margin, combined with the excess capability in the as designed equipments, systems and components provide the potential to institute higher operating power with out major upgrade or modification of the NSSS and BOP hardware.

This approach is based on and is consistent with, the BWR generic extended power uprate guidelines that are presented in NEDC-32424P (Feb, 1995), "Generic Guidelines for General Electric Boiling Water Extended Power Uprate", best known as ELTR1 and NEDC-32523P, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," known as ELTR-2. The staff issued a Safety Evaluation on ELTR-2 in a letter dated May 18, 1998. SNC stated that the generic guidelines were evaluated and found to be applicable to plant Hatch's extended power uprate. The plant specific evaluation for Hatch is presented in GE topical report NEDC-32749P(July 1997), " Edwin 1. Hatch Nuclear Plant Request for License amendment: Extended Power Uprate Operation," and it's Supplements (Enclosure 6 to the August 8, 1997 letter).

II. EVALUATION

2.0 REACTOR CORE AND FUEL PERFORMANCE

2.1 Fuel Design and Operation

The licensee stated that a new fuel design is not needed to achieve the extended power uprate. However, SNC may employ revised loading patterns, larger batch sizes, and potentially new fuel designs in order to attain additional operating flexibility and to maintain fuel cycle length.

The licensee will continue to meet all fuel and core design limits through planned use of fuel enrichment, and burnable poison, supplemented by control rod pattern and core flow adjustment. The proposed extended 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.

2.2 Thermal Limits Assessment

The licensee selected Cycle 14, or reload 13 of Hatch Unit 2 core as a representative core for the power uprate evaluation. SNC performed anticipated operational occurrence (AOO) transient analysis at the proposed extended power level of 2763 MWt. The licensee determined the most limiting AOO transient event and established the corresponding operating limit minimum critical power ratio (OLMCPR) required to assure the regulatory and safety limits are not exceeded for a range of postulated transient events. However, the licensee assumed a safety limit minimum critical power ratio (SLMCPR) of 1.12, which was calculated based on Unit 2 reload 13 core.

SNC proposes that the cycle specific operating and safely limit MCPR be calculated during each reload based on the actual core configuration. In addition, the licensee as part of the upcoming Cycle 15 reload analysis performed a cycle-specific SLMCPR analysis based on the extended power uprate using staff approved methods stated in the GESTR reference document (NEDO-24011P-A). The results indicated that the SLMCPR of 1.12 assumed in the uprate analysis was conservative by a value of 0.01. SNC concluded that the SLMCPR evaluation at the extended power, while not based on the actual core at the uprated power, was representative of the cycle -specific reload licensing calculation and the thermal margin design limits will be maintained. Where extended power uprate results in a greater number of bundles operating near the limit, the SLMCPR may be increased to provide the same statistical confidence level that the rods will avoid boiling transition. Transient events will continue to be evaluated against this SLMCPR, using NRC approved procedures, when establishing the operating limit MCPR.

The licensee performed AOO transient analysis to determine the changes in the MCPR for the postulated transient events based on the assumed SLMCPR of 1.12. The operating limits assure that the safety limits are not exceeded in the event of the limiting postulated transients. Thermal limit such as the average planar linear heat generation rates (APLHGR) ensure that the design margin for the peak cladding temperature limits for the limiting loss-of coolant analysis (LOCA) and fuel mechanical design basis are maintained.

SNC responded (letter to NRC dated March 9, 1998) to the staff's request for additional information (RAI) to confirm that the SLMCPR assumed at the uprated power was conservative. Furthermore, the licensee will perform cycle-specific fuel thermal-mechanical limit evaluations based on actual extended power core configuration during the reload analysis.

2.3 Reactivity Characteristics

All minimum shutdown margin requirements that apply to cold (212 deg.F or less) conditions, will be maintained without change. Operation at higher power could reduce the excess reactivity during the cycle. This loss of reactivity is not expected to significantly degrade the ability to manage the power distribution through the cycle to achieve the rerated power level. The lower reactivity will result in an earlier all-rods-out condition. Any reduction in operational shutdown margins may need to be accommodated through core design. The technical specification requirements for shutdown margin will continue to be met.

2.3.1 Power/Flow Operating Map

The extended power/flow operating map includes the operating domain changes for the extended power. SNC stated Hatch was licensed with extended load limit analysis (ELLA) before and after the original power uprate. The rod line for ELLA plant is approximately 108% of the original power and the rod line maximum ELLA or MELLA is 120% of the original power.

Currently, Hatch is analyzed to operate with core flow between 87% to 105% core flow full power of 2558 MWt. The 87% core flow is consistent with ELLA and corresponds to approximately the 108% rod (flow control) line. For the proposed extended power, the operating range for core flow will be 91% to 105%. The proposed rod line for the extended power uprate corresponds to the 120% rod line relative to the original licensed power and 115% rod line relative to the current 2558 MWt.

The submittal contained the proposed power flow map which indicates the initial licensed power, the 100% power stretch operating line as well as the proposed operating line.

According to the licensee the proposed 91% to 105% core flow rate will be achieved with the 13% power uprate. SNC concluded that all safety analysis for the extended power uprate were performed considering the proposed power-to-flow map.

2.4 Stability

Unit 2 installed digital power range neutron monitor (PRNM) system with an oscillating power range monitor (OPRM) and Hatch Unit 1 will install a similar system in Fall of 1998. Currently, the OPRM for Unit 2 is set to ALARM mode, while the algorithm which are used to detect, suppress and limit cycle oscillations are being validated to the Hatch conditions. However, SNC stated it will arm the OPRM prior ro implementing the extended power uprate for Units 1 and 2.

According to SNC, operation at uprated power will not effect the ability of this detect- and suppress OPRM system to mitigate a stability event. The Option III solution combines closely spaced local power range monitor (LPRM) detectors into cells to effectively detect a core-wide or regional oscillation. Moreover, the licensing methodology used to determine the Option III set points is intended to provide adequate protection for the safety limit minimum critical power ratio (SLMCPR) and the methodology is independent of reactor power. For extended power uprate, the power set point is re-scaled to maintain the same absolute power at the boundary of the enabled (armed) region, but the percent core flow boundary remains unchanged.

2.5 Reactivity Control

2.5.1 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 of 1035 psig with additional 40 psid for the bottom head location. The extended power uprate does not increase the reactor dome pressure in reference to the initial 5% power uprate because the high pressure turbine was modified to accommodate the higher steam flow for the extended power uprate.

The structural and functional integrity of the CRD mechanism has been designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The design pressure of the CRD mechanism correspond to bottom head pressure of 1250 psig during normal operation and 1375 psig (110%) for upset condition. The over pressure transient analysis for the extended power uprate resulted with a bottom head pressure of 1347 psig which remains below the 110% ASME Code allowable peak pressure.

CRD insertion and withdrawal require a minimum pressure differential of 250 psid between the HCU unit and the bottom head pressure. During the original 5% power uprate analysis, the CRD pumps were evaluated against this requirement and were found to have sufficient capacity. SNC stated that since the implementation of the 5% uprate, no visible difference in the CRD system operation occurred. Based on the fact that the dome pressure will not be increased in the current extended power uprate, the CRD mechanism is expected to function with no change in performance. The licensee will continue to monitor, through various plant TS surveillance requirements, the scram time performance to ensure that the original licensing bases for the CRD system are maintained. This approach is consistent with that proposed by GE in the generic references.

3.0 REACTOR COOLANT SYSTEM

3.1 Nuclear System Pressure Relief

The nuclear boiler pressure relief system prevents over pressurization 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 appropriate SRV set points also ensures that adequate differences between operating pressure and set points are maintained (i.e., the "simmer margin"), and that any increase in steam dome pressure does not result in an increase in unnecessary SRV actuation.

The SRV set points were reevaluated to ensure that the ASME mechanical limits and the simmer margin are maintained. The licensee does not intend to increase the operating pressure to achieve the power uprate therefore, the SRV flow rates and setpoints are acceptable.

3.2 Code Over pressure Protection

The results of the over pressure protection analysis are contained in each cycle-specific reload amendment submittal. The design pressure of the reactor pressure vessel (RPV) remains at 1250 psig. The ASME code allowable peak pressure for the reactor vessel is 1375 psig (110% of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event for Hatch is an MSIV closure with a failure of the valve position scram. As part of the extended power uprate over pressure analysis, turbine trip with bypass failure and neutron flux scram was also evaluated and found to be less limiting than the MSIV closure with the failure of direct scram. The MSIV closure was analyzed by the licensee using the NRC approved methods (ODYN), with the following exceptions: (1) the MSIV closure event be analyzed at 102% of the uprated core power and 105% of rated steam flow; (2) the maximum initial reactor dome pressure was assumed to be 1058 psig, which is higher than the nominal uprated pressure (1035 psig); and (3) one SRVs were assumed out-of-service for consistency with previous analyses. The SRVs have an assumed opening tolerance of 3% above the normal set points. The peak dome pressure for the extended power uprate increases to 1325 psig and the corresponding bottom head pressure is 1347 psig, which are below the allowable peak pressure of 1375 psig.

3.4 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. SNC stated that at 105% core flow, the calculated pump, pump motor, and MG Set power requirements remain within the individual equipment rating. The Hatch units are also licensed for increased core flow (ICF) of 105% at 100% of current power (2436 MWt). The licensed core flow is not being increased under the extended power uprate. 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 NPSH but, these increases are within the capability of the equipment.

The cavitation protection interlock will remain the same, 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 due to the higher feedwater flow, therefore, the cavitation interlock can be maintained.

The licensee evaluated the NPSH and stated that at full power, the power uprate does not increase the NPSH significantly nor does it reduce the NPSH margin. The reactor operating pressure will not be increased in reference to the current 5% power uprate.

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 set points.

The licensee concluded that uprated power operation is within the capability of the recirculation system. The licensee reviewed the characteristic pump curves and confirmed the new operating range would be with in the pump design operating range.

3.7 <u>Main Steam Isolation Valves (MSIVs)</u>

The main steam isolation valves (MSIVS) are required to operate with in the TS specified limits at all design and operating conditions upon receipt of a closure signal. The licensee evaluated the MSIVs and concluded that the uprated power conditions does not effect the structural integrity of the MSIVs or the s[illegible] function of the valves. The licensee stated that the closure function and closure timing of the MSIVs would not be effected by the power uprate. The Hatch units evaluation results are consistent with the bases and conclusions of the generic evaluation in Section 4.7 of NEDC-32523P,Supp 1 (ELTR-2).

Performance will be monitored by surveillance requirements in the Technical Specification to ensure original licensing basis for the MSIV's are preserved. The licensee stated that the existing design pressure and temperature bounds the normal operating conditions and review of the over pressure analysis also confirms that the peak pressures remain bounded by the MSIV's design capability.

3.8 <u>Reactor Core Isolation Cooling System (RCIC)</u>

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 a low pressure core cooling system. The submittal stated that the recommendations of GE SIL No. 377 have been implemented on the RCIC system. This modification is intended to achieve the turbine speed control/system reliability desired by SIL 377, and is consistent with the requirements in the staff 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 over speed, high steam flow isolation, low pump suction pressure and high turbine exhaust pressure.

The staff requires that licensee provide assurance that the RCIC system will be capable of injecting their design flow rates at the higher reactor operating pressures associated with extended power uprate. Additionally, the licensee must also provide assurance that the reliability of this system will not be decreased by the higher loads placed on the system or because of any modifications made to the system to compensate for these increased loads.

In the previous 5% power uprate, the licensee evaluated the RCIC system performance for a pressure of 1195 psig, which is the SRV upper set point. For the current extended power uprate, the licensee stated the SRV set point or the RCIC actuation RX pressure would not change and the calculated minimum RCIC injection rate will remain at 400 gpm. The system has also been evaluated for loss of feedwater transient events. The RCIC reliability will be monitored under the Maintenance Rule. The RCIC system evaluation is consistent with the bases and conclusions of the generic evaluation.

3.9 Residual Heat Removal System

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.

3.9.1 Shutdown

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125 deg.F in approximately 20 hours, using two RHR loops. The licensee stated that actual operating experience shows that the actual time is 14 hours. At the uprated power level the decay heat is increased proportionally, thus it will require more time to reach the shutdown temperature. Shutdown cooling calculations performed by the licensee showed that the reactor coolant will reach 125° F in 18 hours. Therefore, the shutdown cooling will take additional 4 hours in the extended power uprate operation..

The licensee also evaluated the shutdown cooling mode's system capability for the extended power uprate with one RHR system in service and with 95° F RHR service water temperature. The results of the analysis showed that the reactor could be cooled to 212° F in less than 36 hours. This evaluation meets the draft Regulatory Guide 1.139 recommendation (212 deg. F reactor fluid temperature) with 1 RHR out of service.

3.9.2 Suppression Pool Cooling and Containment Spray Modes

The Suppression Pool Cooling (SNC) 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 LOCA.

3.9.4 Fuel Pool Cooling Assist Mode

During normal plant shutdown, with the vessel head removed, the RHR system can be aligned to assist the fuel pool cooling and cleanup (FPCC) system to maintain the fuel pool temperature within acceptable limits. The analysis in section 6.3 of the licensee submittal indicates that the fuel pool temperature will remain within limits under power uprate conditions, and therefore the capability of the fuel pool cooling assist mode is acceptable for power uprate. **This is SPLB responsibility.**

3.10 Reactor Water Cleanup(RWCU)System

This is EMCB responsibility

4.0 ENGINEERED SAFETY FEATURES

4.2 Emergency Core Cooling Systems (ECCS)

The HPCI, RHR(LPCI mode), CS and ADS are the ECCS required to provide core cooling in the event of LOCA. The following subsections review the impact of the power uprate on the safety function of the ECCS.

4.2.1 High Pressure Core Injection (HPCI)

The HPCI system safety function is to provide reactor vessel makeup water inventory during small and intermediate break LOCAs. The HPCI system also serves as a back up system for the RCIC if normal feedwater is lost. The system operates over a pressure vessel range of 150 to 1195 psig, with the latter pressure corresponding to the SRV set points.

The licensee stated that the HPCI turbine design pressure and temperature is 1250 psig at 575° F and the HPCI pump design pressure is 1500 psig. The licensee concluded that the extended power uprate operating conditions are bounded by the HPCI system design conditions.
The HPCI is capable of providing the design flow at the power uprate conditions.

The licensee also implemented the GE SIL 480 recommendation prior to the 5% power uprate. SIL 480 recommends that the HPCI system be modified in order to minimize the potential for system trip during startup transient. For the Hatch units, the HPCI system uses a ramp generator during system startup to provide controlled turbine acceleration and this minimizes the control valve cycle during system initiation.

The licensee provided assurance that the reliability of the HPCI system will not be decreased by the higher loads placed on the system or because of any modifications made to this system to compensate for these increased loads. The reliability of the HPCI system will be monitored under the Maintenance Rule. The staff found the evaluation of the HPCI system acceptable.

4.2.2 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 set points 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. 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 for the Hatch units there is no impact due to the power uprate, except for the NPSH available margin which was discussed in Section 4.2 above. This evaluation is acceptable to the staff.

4.2.3 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 set points 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. These systems do not experience different operating conditions due to power uprate, therefore 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 for the Hatch units there is no impact due to the power uprate, except for the NPSH available margin which was discussed in Section 4.2 above. This evaluation is acceptable to the staff.

4.2.4 Automatic Depressurization Systems (ADS)

The ADS uses safety/relief valves to reduce reactor pressure following a small break LOCA with HPCI failure. This function allows low pressure coolant injection (LPCI) and core spray (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.

4.3 ECCS Performance Evaluation

The emergency core cooling systems (ECCS) are designed to provide protection against hypothetical loss-of-coolant accidents (LOCAs) caused by ruptures in the primary systems 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 licensee used the staff approved SAFER/GESTR (S/G) methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria. The ECCS-LOCA evaluation for the extended power uprate was documented in the GE document NEDC-32720P (March 1997) and results were discussed in this section.

In the ECCS-LOCA analysis, the licensee assumed the highest power rod in the peak bundle to be at the peak linear heat generation rate. In the current submittal, the licensee stated the fuel type will not be changed and the fuel parameters would remain constant. Therefore, in this evaluation the PLHGR will not be changed and higher core power distribution would alter the average bundle power, but will not have significant effect on the PCT. According to SNC, the licensing basis PCT changed from 1686° F to 1688° F for power of 2558 MWt and 2763 MWt respectively.

The result of the licensee's ECCS performance evaluation showed that the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K are satisfied for the extended power uprate. A sufficient number of plant-specific break sizes were evaluated to establish the behavior of both the nominal and Appendix K PCT as a function of break size. Different single failures were also investigated in order to clearly identify the worst cases. The HATCH specific analysis was performed with a conservatively high Peak Linear Heat Generation Rate (PLHGR) and a conservative Minimum Critical Power Ratio (MCPR). In addition, some of the ECCS parameters were conservatively established relative to actual measured ECCS performance. The analysis also meets the other acceptance criteria of 10 CFR 50.46. Compliance with each of the elements of 10 CFR 50.46 is documented in the GE Licensing Topical Report for the HATCH units. The results for the limiting break and single failure (the design-basis-accident (DBA)), for the limiting GE13 fuel are presented below. The nominal PCT is 1133 deg.F, the licensing basis PCT is 1688 deg.F, the Appendix K PCT is 1664 deg.F, and the upper bound PCT is 1464 deg.F. These temperatures meet the requirements of the approved SAFER/GESTR-LOCA methodology stated below. The SAFER/GESTR-LOCA methodology requires:

- The Licensing Basis PCT (LBPCT) must be less than 2200 deg.F. This LBPCT is derived by adding appropriate margin for specific conservatism required by Appendix K to the limiting PCT value calculated using nominal inputs, the nominal PCT (NOMPCT).
- The Upper Bound PCT (UBPCT) must be less than the LBPCT. The UBPCT is the estimated mean of the PCT distribution for the limiting LOCA plus the estimated standard deviation of the distribution of PCTs for the limiting case LOCA. The UBPCT calculated in this way is presumed to bound the 95th percentile of the PCT distribution for the limiting case LOCA, and for all LOCAs within the design basis.

- The UBPCT is less than the LBPCT when the limiting nominal PCT is lower than 1600 deg. F. Therefore, it is required that the UBPCT be below 1600 deg.F; otherwise, additional plant specific analyses must be done.

A 0.85 MAPLHGR multiplier will be utilized for single loop operation as previously accepted by the staff. The previously multipliers are conservative with respect to the SAFER/GESTR-LOCA results because the S/G model results in more efficient heat removal during the boiling transition phase than the previous evaluation model used to derive these multipliers. The licensee stated that single loop operation (SLO) is limited to 88% of the current power level. At uprated conditions, this corresponds to 83.8% of the uprated power level. The licensee provided further assurance by letter dated **xxxx xx, xxxx** 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, Hatch Units I and 2 will continue to meet the NRC LOCA licensing analysis and results requirements. The licensee will evaluate and verify the acceptability of the results of the plant specific LOCA analysis at each reload.

6.5 Standby Liquid Control System (SLCS)

The function of the SLCS is to provide the capability of bringing the reactor from full power to a cold xenon-free shutdown assuming that none of the withdrawn control rods can be inserted.. SLCS shutdown capability (boron concentration) is reevaluated for each fuel reload to ensure sufficient shutdown margin is available.

The SLCS is designed for injection at a maximum reactor pressure equal to the minimum SRV set point pressure. The nominal SRV set points and operating pressure will not be changed for the Hatch extended power rerate. The SLCS pumps are positive displacement pumps, where the small pressure increase related to the 3% tolerance on the as-found SRV opening pressure does not affect the rated flow to the reactor. Therefore, the capability of the SLCS to provide its backup shutdown function is not affected by power rerate. Also, because there is no increase in system operating pressure, there is no reduction in the SLCS pump relief valve pressure margin, or in the pump motor horsepower requirements. The SLCS performance is evaluated in Section 9.3.1 for a representative core design.

9.0 REACTOR SAFETY PERFORMANCE FEATURES

The staff requested that the licensee identify all codes/methodologies used to obtain safety limits and operating limits and how they verified that these limits were correct for the appropriate uprated core. The licensee was also requested to identify and discuss any limitations associated with these codes/methodologies that may have been imposed by the staff. In a letter dated March 9, 1998, the licensee responded to this staff request. The licensee stated that the restrictions and conditions applicable to GENE's core and fuel design are documented in GESTAR II, NEDE-24011-P-A-13, Revision 13, "General Electric Standard Application for Reactor Fuel", August 1996 and GESTAR II, NEDE-24011-P-A-13-US, Revision 13, "General Electric Standard Application for Reactor Fuel (Supplement for the United States)," August 1996. The approved codes and methodologies are specified in this document. The licensee stated that the evaluations were performed and verified by a third party prior to submittal to the NRC. The licensee stated that the limitations, restrictions, and conditions specified in the approving NRC safety evaluations were adhered to when applying the codes to the power uprate analyses. This is acceptable to the staff.

9.1 Reactor 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 used the NRC-approved methodologies outlined in the generic report (NEDC-32424P, Table E-1) to establish the transient events to be analyzed for the extended power uprate, the power level to assume and the computer model to use. SNC analyzed the following transient events and Table 9-2 of Enclosure 6 provides the transient results.

(1) Turbine Trip with Bypass Failure

(2) Generator Load Rejection with Bypass Failure

(3) Feedwater Controller Failure:
Max Demand
Max Demand with Bypass Failure

(4) Loss of Feedwater Heating

(5) Rod Withdrawal Error

(6) Slow Recirculation Flow Increase

The licensee selected Hatch Unit 2 for the reactor transient analysis since the two units have similar vessel size, core power and SRV set points. The Unit 2, reload 13 core served as the bounding representative core and all the analyses were performed at full extended power at the maximum allowed core flow. The safety limit minimum critical power ratio (SLMCPR) was assumed to remain the same as Unit 2 reload 14 value and the licensee stated the SLMCPR will be evaluated for the specific core designed for the extended uprate consistent with Section 3.4 of NEDC-32523P. For all the transients, the licensee assumed one SRV out service and direct or the statistical allowance for 2% power uncertainty was included in the analysis.

The licensee analyzed the sensitivity of each limiting transient category to core flow, feedwater temperature and cycle exposure, The limiting transient analysis results for the extended power uprate are summarized in Table 9.2 of the licensee submittal (NEDC-32749P) The licensee stated that there were no changes to the mitigation trip set points for the pressurization events and the basic characteristic of the transient events did not change with power uprate. However, due to the higher decay heat for the extended power, the automatically actuated system will require slightly more time to restore the water level.

Operator action is only necessary for long term plant shutdown once the water level is restored and no new operator action or shorter response time is needed for the extended power uprate. The licensee also stated that multipliers for off-rated MCPR and MAPLHGR will be re-evaluated during core reload analysis. The power dependent MCPR and MAPLHGR will provided the basis for instrumentation set points. The power uprate analysis used the staff approved GEMINI methodology for 100% initial power and REDY analysis for the 102% initial power. The analysis plan proposed by the licensee is acceptable. The staff will verify the acceptability of the results when each reload document is submitted.

9.2 Design Basis Accidents:

(This section is not under SRXB review).

9.3 SPECIAL EVENTS

9.3.1 Anticipated Transients Without Scram (ATWS)

A generic evaluation of the ATWS event is presented in NEDC-32424P. This evaluation concludes that the ATWS acceptance criteria for fuel, reactor pressure vessel (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 20%; dome pressure increase is equal to or less than 1080 psig; the lowest SRV opening set point up to the TS analytical limit is 1195; ATWS high pressure set point up to the analytical limit is 1220 psig; the SRV capacity must be greater than 76 % of the initial steam flow rate at 1195 psig opening set point; and equivalent boron injection 86 gpm is available. Based on the analysis in NEDC-32424P, Hatch meets most of the bounding plant parameters, however the SRV capacity is lower. The licensee performed plant specific ATWS analyses at the extended power uprate.

9.3.2 Station Blackout

Plant response and coping capabilities for a station blackout (SBO) 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 set points. There are no changes to the systems and equipment used to respond to an SBO, nor is the coping time changed. The plant Hatch coping time for SBO event is four hours.

The following areas contain equipment necessary to mitigate the SBO event: Control Room; RCIC and HPCI Equipment Room; Steam Pipe Chase/Steam Tunnel; Drywell and Suppression Pool; and RHR Corner Room.

The licensee stated none of the areas will experience any increase in normal temperatures due to the power uprate and following an SBO event equipment necessary for event mitigation will not effected. Assuming the suppression pool cooling was initiated after one hour into the SBO event, when the alternate AC is assumed available, the peak pool temperature is 167 F. If the SBO is initiated four hours later, the peak pool temperature is 194 F. This acceptable temperature for containment and for the ECCS pump operation.

Besides the increased heat load effects, an increase in the vessel make up water system is required for extended power uprate. SNC analysis showed that 77,000 gallons of make up water inventory was required for the 13% power uprate condition during the four hour SBO coping period. The condensate storage tank is designed to provide 100,000 gallons of make inventory for isolation invents. Therefore, adequate water volume is available for SBO event for 2763 MWt uprate reactor operation. Based on the above evaluation and assurances by the licensee, the SBO coping capabilities are not adversely affected by power uprate and are acceptable to the staff.

III. CONCLUSION

The licensee has provided analytical evaluations, evaluations of components and equipment, and a discussion of the impact of power uprate on licensing criteria to demonstrate the feasibility of an extended power uprate of 8%. The proposed power uprate represents a 8% increase from the current power and a 13% power increase from the original licensed power of 2436 MWt. Based on the staff review of the information submitted in support of the power uprate of 8%, the staff found the power uprate to be acceptable and to meet the required regulations. Operation of the Hatch units at a power uprate of 8% above the current power level (13% above the original licensed power)will not cause unacceptable consequences to the health and safety of the public.

REFERENCES

THE REFERENCE SECTION SHOULD BE PREPARED BY THE PM. REFERENCES ARE INDICATED IN THE TEXT AS NEDC REPORTS AND LETTERS.