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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 214 TO FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 OVERVIEW

1.1 Introduction

By letter dated August 8, 1997 (Reference 1), as supplemented by letters dated March 9, 1998, (Reference 2), May 6, 1998 (Reference 3), July 6, 1998 (Reference 4), July 31, 1998 (Reference 5), September 4, 1998 (Reference 6), and September 11, 1998 (Reference 7), Southern Nuclear Operating Company, Inc. (SNC, the licensee), et al., proposed license amendments to change Facility Operating License Nos. DPR-57 and NPF-5 for the Edwin I. Hatch Nuclear Plant (Hatch), Units 1 and 2. Advance information was provided by letter dated April 17, 1997 (Reference 8). The proposed changes would increase the maximum licensed thermal power level by 8 percent, from the current limit of 2558 megawatts thermal (MWt) to 2763 MWt. The amendments would also approve changes to the Technical Specifications appended to the operating licenses to implement uprated power operation.

1.2 Background

Hatch Units 1 and 2 are currently licensed to operate at a maximum reactor power level of 2558 MWt . The licensee, in conjunction with General Electric Company (GE), undertook a program to uprate the maximum reactor power level by 8 percent to 2763 MWt.

The licensee's plant-specific engineering evaluations supporting the extended power uprate were performed in accordance with guidance contained in the GE licensing topical report (ELTR-1) NEDC-32424P, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," dated February 1995 (Reference 9). This topical report was previously reviewed and endorsed by the NRC staff in a staff position paper dated February 8, 1996 (Reference 10). Generic evaluations performed to support the extended power uprate are addressed in GENE ELTR NEDC-32523P, "Generic Evaluations for General Electric Boiling Water Reactors Extended Power Uprate" (Reference 11). The staff has reviewed NEDC-32523P concurrently with the power uprate amendment request submitted for the Monticello Nuclear Generating Plant and the staff's safety evaluation was issued on September 14, 1998 (Reference 12).

The licensee's submittal was reviewed with consideration given to the recommendations from the report of the Maine Yankee Lessons Learned Task Group. This report is documented in SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," dated February 18, 1997 (Reference 13). The Task Group's main findings centered around the use and applicability of the computer codes and analytical methodologies used for power uprate evaluations. The Task Group also recommended that a standard review procedure for power uprate be developed to ensure that all appropriate review areas are addressed. For the boiling water reactor (BWR) extended power uprate program, the staff had previously established review criteria and acceptable computer codes and analytical methodologies used for power uprate evaluations. In light of the Task Group's recommendations, the staff has expanded the review criteria to include areas such as human performance and offsite power stability. As a result, the staff concludes that the Maine Yankee Lessons Learned recommendations were appropriately considered in the review of the Hatch extended power uprate request.

1.3 Approach

The proposed extended power uprate is accomplished by extending the power-to-flow map within (approximately) equivalent rod and core flow control lines. The proposed rod and flow control line for the 13 percent power increase corresponds to the 120 percent rod and flow control line relative to the original licensed power of 2436 MWt and approximately 115 percent rod and flow control line relative to the original 5 percent increase in power to 2558 MWt. The licensee also stated that the proposed rod and flow control line is consistent to the maximum extended load line limit analysis (MELLLA) 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 MELLLA plant at the original licensed power.

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

SNC proposed to achieve the extended power uprate by supplying the higher steam flow to the turbine generator. The licensee had modified the high pressure turbine to accommodate the higher steam supply. SNC also states that improvements in the analytical techniques (i.e., computer codes and data) for several decades of BWR safety technology, plant performance feedback, and improved fuel and core design have resulted in a 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 without major upgrade or modification of the nuclear steam supply system (NSSS) and balance of plant (BOP) hardware.

2.0 EVALUATION OF SYSTEMS, STRUCTURES, AND COMPONENTS

The staff's review of the Hatch extended power uprate amendment request used applicable rules, regulatory guides, Standard Review Plan (SRP) sections, and NRC staff positions regarding the topics being evaluated. Additionally, the submittal was evaluated for compliance with the generic BWR power uprate program. Detailed discussions of individual review topics follow.

2.1 Reactor Core and Fuel Performance

2.1.a 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.

2.1.b Thermal Limits Assessment

The licensee selected Cycle 14, or reload 13, of the Hatch Unit 2 core as a representative core for the extended power uprate evaluation. SNC performed anticipated operational occurrence (AOO) transient analyses 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 (MCPR) 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 limit MCPR and SLMCPR 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 analyses 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 limits, such as the average planar linear heat generation rates, ensure that the design margin for the peak cladding temperature limits for the limiting loss-of-coolant accident (LOCA) and fuel mechanical design basis are maintained.

SNC, in Reference 2, confirmed that the SLMCPR assumed at the extended uprated power was conservative. Furthermore, the licensee will perform cycle-specific fuel thermal-mechanical limit evaluations based on the actual extended power core configuration during the reload analysis. Therefore, the staff concludes that the thermal limits are acceptable for the extended power uprate.

2.1.c Reactivity Characteristics

All minimum shutdown margin requirements that apply to cold (212 °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 uprated power level. The lower reactivity will result in an earlier all-rods-out condition. The Technical Specifications requirements for shutdown margin will continue to be met. Therefore, the staff concludes that the reactivity characteristics are acceptable for the extended power uprate.

Power/Flow Operating Map

The extended power/flow operating map includes the operating domain changes for the extended power uprate. Currently, Hatch is analyzed to operate with core flow between 87 percent to 105 percent core flow at 2558 MWt. The 87 percent core flow is consistent with the extended load line limit analysis (ELLLA) and corresponds to approximately the 108 percent rod (flow control) line. For the proposed extended power uprate, the operating range for core flow will be 91 percent to 105 percent. The proposed rod line for the extended power uprate corresponds to the 120 percent rod line relative to the original licensed power and 115 percent rod line relative to the current 2558 MWt. Therefore, the rod line for the extended power uprate is consistent with the maximum ELLLA rod line of the original rated power of 2436 MWt.

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

According to the licensee, the proposed 91 percent to 105 percent core flow rate will be achieved with the extended power uprate. SNC confirmed that all safety analyses for the extended power uprate were performed considering the proposed power-to-flow map. The staff concludes that the proposed power-to-flow map is acceptable.

2.1.d Stability

Unit 1 and Unit 2 installed digital power range neutron monitor systems with an oscillating power range monitor (OPRM). Currently, the OPRM for both units is set to ALARM mode, while the algorithms that 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 to implementing the extended power uprate for Units 1 and 2.

According to SNC, operation at extended uprated power will not affect 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 detectors into cells to effectively detect a core-wide or regional oscillation. Moreover, the licensing methodology used to determine the Option III setpoints is intended to provide adequate protection for the SLMCPR and the methodology is independent of reactor power. For extended power uprate, the power setpoint 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. Therefore, the staff concludes that the units will be able to mitigate a stability event for the extended uprated power level.

2.1.e Reactivity Control

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 pounds per square inch gauge (psig) with additional 40 pounds per square inch differential (psid) for the bottom head location. The extended power uprate does not increase the reactor dome pressure in reference to the initial 5 percent 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 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III (See Section 3.2). The design pressure of the CRD mechanism corresponds to bottom head pressure of 1250 psig during normal operation and 1375 psig (110 percent) for upset condition. The overpressure transient analysis for the extended power uprate resulted in a bottom head pressure of 1347 psig which remains below the 110 percent ASME Code allowable peak pressure.

CRD insertion and withdrawal require a minimum pressure differential of 250 psid between the hydraulic control unit and the bottom head pressure. During the original 5 percent 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 percent 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. Therefore, the staff concludes that reactivity control is acceptable for the extended power uprate.

2.2 Reactor Coolant System

2.2.a Nuclear System Pressure Relief

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 at the higher steam flow condition corresponding to uprated power. The appropriate SRV setpoints also ensure that adequate differences between operating pressure and setpoints 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 setpoints 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 extended power uprate; therefore, the SRV flow rates and setpoints are acceptable for the extended uprated power.

2.2.b 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 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 for Hatch is a main steam isolation valve (MSIV) closure with a failure of the valve position scram. As part of the extended power uprate overpressure 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 was analyzed at 102 percent of the uprated core power and 105 percent 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 SRV was assumed to be out of service for consistency with previous analyses. The SRVs have an assumed opening tolerance of 3 percent above the nominal setpoints. The peak dome pressure for the extended power uprate increases to 1325 psig and the corresponding bottom head pressure is 1347 psig. These pressures are below the allowable peak pressure of 1375 psig. The results of the licensee's analysis are acceptable.

2.2.c Reactor Recirculation System

The extended 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 the extended uprated power determined that the core flow can be maintained with only a slight increase in pump speed.

The Hatch units are also licensed for increased core flow of 105 percent at 100 percent of current power (2558 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 extended uprate condition will increase the power demand of the recirculation pump 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, 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 extended 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 extended power uprate does not increase the NPSH significantly nor does it reduce the NPSH margin. The reactor operating pressure will not be increased from its current level.

The recirculation drive flow stops were reviewed by the licensee for application to extended uprated power conditions. Since extended 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 the extended 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 within the pump design operating range. The staff concludes that the recirculation system is acceptable for the extended power uprate.

2.2.d Main Steam Isolation Valves

The MSIVs are required to operate within 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 extended uprated power conditions do not affect the structural integrity of the MSIVs or the safety function of the valves. The licensee stated that the closure function and closure timing of the MSIVs would not be affected by the extended power uprate. The Hatch units' evaluation results are consistent with the bases and conclusions of the generic evaluation in Section 4.7 of Reference 11.

Performance will be monitored by surveillance requirements in the TS to ensure that the original licensing basis for the MSIVs is preserved. The licensee stated that the existing design pressure and temperature bounds the normal operating conditions and review of the overpressure analysis also confirms that the peak pressures remain bounded by the MSIVs' design capability. The staff concludes that the MSIVs are acceptable for the extended power uprate.

2.2.e Reactor Core Isolation Cooling System (RCIC)

The RCIC provides core cooling when the 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 licensee's submittal stated that the recommendations of GE Service Information Letter (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 safety evaluation (SE) of the generic topical report. The purpose of the modification is to mitigate the concern that a slightly higher steam 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.

In the original 5 percent power uprate, the licensee evaluated the RCIC system performance for a pressure of 1195 psig, which is the SRV upper setpoint. For the proposed extended power uprate, the licensee stated the SRV setpoint and the RCIC initiation setpoint 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 and is acceptable.

2.2.f Residual Heat Removal System (RHR)

The RHR system 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 LPCI mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The effects of the extended power uprate on these operating modes are discussed in the following paragraphs.

2.2.f.1 Shutdown

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125 °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 extended 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, which is an additional 4 hours. The staff finds this acceptable since it is still within the objective of 20 hours.

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 °F reactor coolant temperature) with one RHR loop out of service and is acceptable.

2.2.f.2 Suppression Pool Cooling and Containment Spray Modes

The Suppression Pool Cooling (SPC) and Containment Spray 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. Therefore, the SPC and Containment Spray modes are acceptable for extended power uprate.

2.2.f.3 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 in maintaining 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 extended power uprate conditions, and therefore the capability of the fuel pool cooling assist mode is acceptable for the extended power uprate.

2.2.g Reactor Water Cleanup (RWCU) System

The RWCU system removes solid and dissolved impurities from recirculated reactor coolant, thus, reducing the concentration of radioactive and corrosive species in the reactor coolant. The licensee has stated that there will be a slight increase in the iron concentration and conductivity of the reactor water at the extended uprated power level. The staff has reviewed the licensee's assessment that concludes (1) that these differences are not significant and (2) that the RWCU system is capable of performing its function at the extended uprated level, and finds it acceptable.

2.3 Emergency Core Cooling Systems (ECCS)

The high pressure coolant injection (HPCI), RHR (LPCI mode), Core Spray (CS) and automatic depressurization system (ADS) are the ECCS required to provide core cooling in the event of a LOCA. The following subsections review the impact of the power uprate on the safety function of the ECCS.

2.3.a High Pressure Coolant Injection

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 system 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 setpoints.

The licensee stated that the HPCI turbine design pressure and temperature are 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 licensee also implemented the GE SIL 480 recommendation prior to the 5 percent 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 cycling during system initiation.

The licensee provided assurance that the reliability of the HPCI system will not be affected by the extended power uprate operating conditions since there is no change in either the SRV setpoint pressure or the vessel operating pressure. The reliability of the HPCI system will be monitored under the Maintenance Rule. The staff finds the evaluation of the HPCI system to be acceptable.

2.3.b Low Pressure Core Injection System (LPCI mode of RHR)

The hardware for the low pressure portions of the RHR system is not affected by the extended power uprate. The upper limit of the LPCI injection setpoints will not be changed for the extended power uprate; therefore, this system 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 the extended power uprate, there is no impact due to the extended power uprate. The licensee stated that for the Hatch units there is no impact due to the extended power uprate, except for the NPSH available margin, which is discussed in Section 2.8. This evaluation is acceptable.

2.3.c Core Spray System

The hardware for the core spray system is not affected by extended power uprate. The upper limit of the CS injection setpoints will not be changed for extended power uprate; therefore, this system 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 the extended power uprate; therefore, there is no impact due to the extended power uprate. Also, the impact of the extended power uprate on the long-term flow 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 extended power uprate, except for the NPSH available margin, which is discussed in Section 2.8. This evaluation is acceptable.

2.3.d Automatic Depressurization System (ADS)

The ADS uses SRVs 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 the extended power 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 the extended power uprate.

2.4 ECCS Performance Evaluation

The ECCS are designed to provide protection against postulated 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 Part 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 results of the ECCS-LOCA evaluation for the extended power uprate were documented in the GE document NEDC-32720P (March 1997) and are 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 (PLHGR). In the current submittal, the licensee stated that a new fuel type is not needed and that the fuel parameters will remain unchanged in the near-term. Therefore, in this evaluation the PLHGR will not be changed and higher core power distribution will alter the average bundle power, but will not have significant effect on the peak clad temperature (PCT). According to SNC, the licensing basis PCT changed from 1686 °F to 1688 °F for the power change from 2558 MWt to 2763 MWt.

The results of the licensee's ECCS performance evaluation showed that the requirements of 10 CFR 50.46 and 10 CFR Part 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 PLHGR and a conservative 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 °F, the licensing basis PCT is 1688 °F, the Appendix K PCT is 1664 °F, and the upper bound PCT is 1464 °F. These temperatures meet the requirements of the approved SAFER/GESTR-LOCA methodology stated below. The SAFER/GESTR-LOCA methodology provides that:

- The Licensing Basis PCT (LBPCT) must be less than 2200 °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 NOMPCT is lower than 1600 °F. Therefore, it is required that the UBPCT be below 1600 °F; otherwise, additional plant-specific analyses must be done.

A 0.85 maximum average planar linear heat-generation rate (MAPLHGR) multiplier will be utilized for single-loop operation as previously accepted by the staff. The previous 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 is limited to 88 percent of the current power level. At extended uprated conditions, this corresponds to 83.8 percent of the extended uprated power level. The licensee provided assurance that the extended 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 requirements. The licensee will evaluate and verify the acceptability of the results of the plant-specific LOCA analysis at each reload.

2.5 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 setpoint pressure. The nominal SRV setpoints and operating pressure will not be changed for the Hatch extended power uprate. The SLCS pumps are positive displacement pumps, where the small pressure increase related to the 3 percent 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 extended power uprate. 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 (Reference 1) for a representative core design.

2.6 Reactor Safety Performance Features

The staff requested that the licensee identify all codes and methodologies used to obtain safety limits and operating limits and asked how the licensee 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 and methodologies that may have been imposed by the staff. In Reference 2, 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 NRC's safety evaluations were adhered to when applying the codes to the extended power uprate analyses. The staff finds that the licensee's actions are acceptable.

2.6.1 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 that are applicable to Hatch and 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 to Reference 1 provides the transient results.

(1) Turbine Trip with Bypass Failure

(2) Generator Load Rejection with Bypass Failure

(3) Feedwater Controller Failure:
Maximum Demand
Maximum 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 setpoints. 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 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 of service and included in the analysis direct or statistical allowance for 2 percent power uncertainty.

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 Reference 1. The licensee stated that there were no changes to the mitigation trip setpoints for the pressurization events and the basic characteristic of the transient events did not change with the extended 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 reevaluated during core reload analysis. The power-dependent MCPR and MAPLHGR will provide the basis for instrumentation setpoints. The extended power uprate analysis used the staff-approved GEMINI methodology for 100 percent initial power and REDY analysis for the 102 percent 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.

2.7 Special Events

2.7.1 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, RPV, and the containment integrity will not be violated for the extended power uprate if the following are met: reactor power increase is equal to or less than 20 percent; dome pressure increase is equal to or less than 1080 psig; the lowest SRV opening setpoint up to the TS analytical limit is 1195; ATWS high pressure setpoint up to the analytical limit is 1220 psig; the SRV capacity must be greater than 76 percent of the initial steam flow rate at 1195 psig opening setpoint; and equivalent boron injection of 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. Therefore, the licensee performed plant-specific ATWS analyses at the extended power uprate power level which the staff found acceptable.

2.7.2 Station Blackout

Plant response and coping capabilities for a station blackout (SBO) event are impacted by operation at the extended uprated power level due to increase in decay heat. 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 4 hours.

The following areas contain equipment necessary to mitigate the SBO event: Control Room; RCIC and HPCI Equipment Rooms; 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 extended power uprate, and following an SBO event, equipment necessary for event mitigation will not be affected. Assuming suppression pool cooling was initiated after 1 hour into the SBO event when the alternate ac power is assumed available, the peak pool temperature is 167 °F. If the SBO is initiated 4 hours later, the peak pool temperature is 194 °F. This is an acceptable temperature for containment and for the ECCS pump operation.

Besides the increased heat load effects, there is an increase in the condensate water requirements for the vessel makeup for the extended power uprate. SNC's analysis showed that 77,000 gallons of makeup water inventory were required for the 13 percent extended power uprate condition during the 4-hour SBO coping period. The condensate storage tank is designed to provide 100,000 gallons of makeup inventory for isolation events. Therefore, adequate water volume is available for SBO event for 2763 MWt extended power uprate reactor operation. Based on the preceding evaluation, the SBO coping capabilities are not adversely affected by the extended power uprate and are acceptable.

2.8 Net Positive Suction Head

By letter dated July 6, 1998 (Reference 4), the licensee responded to the staff's request for additional information (RAI) concerning NPSH. The RAI addressed the Hatch Unit 1 NPSH analysis for ECCS pumps and the required use of containment overpressure to ensure adequate NPSH to the ECCS pumps at the extended power uprate with the new ECCS suction strainers installed with the design loads from Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996. The staff notes that Hatch Unit 2 does not require the use of containment overpressure to ensure adequate NPSH for the ECCS and containment spray pumps. In its RAI response, SNC demonstrated that 2.1 psig of containment overpressure was required to ensure adequate NPSH to the Hatch Unit 1 RHR pumps and 2.0 psig was required for the Hatch Unit 1 core spray pumps during the long-term post-LOCA. Additionally, SNC requested an additional 2.1 psig (5 feet) of containment overpressure above the required to address potential future issues. The following review evaluates the use of containment overpressure at the extended power uprate with the new suction strainers installed.

2.8.1 RHR and CS NPSH Calculations

The NPSH analyses included the head loss across the new ECCS suction strainers, which were installed to meet the requirements of NRC Bulletin 96-03. By letter dated November 5, 1997, SNC provided the final dimensions of the strainers, the head loss across the new strainer with the calculated debris loading (and the basis for the head loss determination), and the resultant NPSH margin with the new strainers installed for Hatch Unit 1. In its submittal, SNC stated that the new strainers were designed to provide adequate NPSH margin with a debris loading greater than the debris loading documented in NRC SE dated June 17, 1997 (Reference 14). SNC designed the Hatch Unit 1 strainer with debris loading described below:

fibrous debris	300 ft^3 (700 lb)
suppression pool sludge	450 lb
dust and dirt	150 lb
epoxy coating	71 lb
unqualified coatings	120 lb
potentially degraded coatings	200 lb
inorganic zinc	47 lb
top coated inorganic zinc	85 lb

The staff notes that changing the debris loading on the strainer changes the calculated head loss across the strainers affecting the NPSH analysis. Therefore, the current NPSH analyses incorporate the ECCS strainer debris loading and head loss discussed above. The debris loadings are in excess of the loads approved by the staff in its June 17, 1997, SE.

The licensee provided evaluations of post-LOCA NPSH for RHR and CS pumps. The evaluations were divided into two portions as follows:

Short-Term: 0 to 600 seconds (10 minutes), no operator action credited, vessel injection phase

Long-Term: 600 seconds to completion of event, operator actions credited, containment cooling phase

2.8.2 Short-Term NPSH Requirements

For short-term operation, the RHR and CS pumps are assumed to be at runout conditions. No attempt to throttle the pumps is made by the operators, therefore, the RHR and CS pumps run at the highest flow rate that piping friction losses and reactor pressure will physically allow. For RHR, runout flow is assumed to 10,600 gpm, and for CS, runout is assumed to be 5900 gpm. The reactor is assumed to be at 0 psig. In accordance with GE calculations for the extended power uprate (2763 MWt), the maximum suppression pool temperature at 10 minutes is 161°F. For conservatism, a suppression pool temperature of 165°F was used in the NPSH calculations.

Based on the preceding assumptions, the licensee evaluated the NPSH Available (NPSHA) using the following equation.

$$NPSHA = \frac{(P_1 - P_{SAT})144}{\rho} + Z - (h_{L_{PIPING}}) - (h_{L_{STRAINER}})$$

where: P_1 = atmospheric pressure, psia
P_{sat} = saturation pressure at suppression pool temp, psia
ρ = density of suppression pool water, lb/ft^3
Z = static head, ft
$h_{Lpiping}$ = piping friction losses, ft
$h_{Lstrainer}$ = strainer head loss, ft

The licensee's calculations demonstrate that at the assumed runout flows and calculated ECCS strainer head loss at 165 °F suppression pool temperature, containment overpressure is not required for both the RHR and CS pumps during the short-term post-LOCA. The staff finds this acceptable at the extended power uprate power level.

2.8.3 Long-Term NPSH Requirements

For long-term operation, greater than 10 minutes post-LOCA, the operators can throttle the RHR and the CS pumps to the design flow rates of 7700 gpm and 4725 gpm, respectively. In accordance with GE calculations for the extended power uprate, the maximum suppression pool temperature during long-term operation is approximately 207 °F, which is used to calculate the NPSH available at peak suppression pool temperature. Using the preceding equation, the licensee demonstrated that 2.1 psig (5 feet) of containment overpressure are required to ensure adequate NPSH to the Hatch Unit 1 RHR pumps and 2.0 psig (4.88 feet) are required for the Hatch Unit 1 core spray pumps at a peak suppression pool temperature of 207 °F. Using the actual suppression pool temperature profile, containment overpressure is needed beginning approximately 3 hours to 17 hours after the LOCA.

Additionally, SNC requested an additional 2.1 psig (5 feet) of containment overpressure above the required 2.1 psig at the peak suppression pool temperature to address potential future issues. This additional containment overpressure margin would be needed from 1.5 hours to approximately 26.5 hours post-LOCA. The requested and required containment overpressures for the RHR and CS pumps were presented in Figures 60-1 and 60-2 in the licensee's July 6, 1998, submittal. The staff notes that the requested additional containment overpressure is less than half of the calculated minimum containment pressure available. However, the licensee did not identify the future issues that could result in needing more containment overpressure than already required.

By letter dated July 31, 1998 (Reference 5), the licensee provided Tables I1-1 and I1-2, which presented a time history of the DBA LOCA minimum pressures as illustrated in Figures 60-1 and 60-2 of the July 6, 1998, submittal. The column headings defined the applicable pressures.

> **Containment Overpressure Available (psi)** - the containment pressure calculated utilizing the minimum containment pressure analysis presented in response to NRC Question 59, submitted July 6, 1998.
>
> **Containment Overpressure Calculated Minimum (psi)** - The amount of containment overpressure available required {sic} to assure adequate NPSH. A negative number in this column indicates that adequate NPSH is available without containment pressure present.
>
> **Containment Overpressure Additional Margin (psi)** - The amount of containment overpressure available with the requested overpressure margin.

These tables adequately describe the minimum containment overpressure available, the containment overpressure required to ensure adequate NPSH, and the requested additional containment overpressure margin at the associated time of the LOCA.

Additionally, in the July 31, 1998, submittal, the licensee made a commitment to notify the NRC if future issues, singularly or collectively, require SNC to take credit for 1 foot (approximately 0.4 psig) of the requested additional containment overpressure margin. For example, the licensee would notify the staff if a change to the RHR NPSH calculation for the long-term resulted in an increase in the containment overpressure required from 2.1 psig to 2.5 psig (5 feet to 6 feet). This commitment is also applicable to the long-term CS NPSH analysis. The staff believes that the proposed notification process would allow the staff an opportunity to evaluate any change that caused a significant increase in the reliance on containment overpressure.

Based on the preceding analysis, the staff finds that with credit for containment overpressure as specified in Table I1-1 and I1-2, NPSH for the ECCS pumps will be available to meet the long-term worst-case scenario. Additionally, the staff concludes that the requested additional containment overpressure margin, 4.2 psig (10 feet) total at a peak suppression pool temperature of 207 °F, is acceptable based on the licensee's commitment to notify the staff of any individual or collective increase of 1 foot of containment overpressure required in the NPSH analyses. The licensee, in accordance with 10 CFR 50.59, would have to submit another license amendment request for any increase in containment overpressure reliance above 4.2 psig at a peak suppression pool temperature of 207 °F.

2.9 Containment Response

2.9.1 Containment System Performance

The Hatch Final Safety Analysis Reports (FSARs) provide the results of analyses of the containment response to various postulated accidents that constitute the design basis for the containment. Operation with 8 percent extended power uprate from 2558 MWt to 2763 MWt would change some of the conditions and assumptions of the containment analyses. Topical Report NEDC-32424 "Generic Guidelines For General Electric Boiling Water Reactor Extended Power Uprate," Section 5.10.2, requires the extended power uprate applicant to show the acceptability of the effect of the uprated power on containment capability. These evaluations will include containment pressures and temperatures, LOCA containment dynamic loads, safety-relief valve containment dynamic loads, and subcompartment pressurization. Appendix G of NEDC-32424 prescribes the generic approach for this evaluation and outlines the methods and scope of plant-specific containment analyses to be done in support of extended power uprate. Appendix G states that the applicant will analyze short-term containment pressure and temperature response using the GE M3CPT code (current analyses). These analyses will cover the response through the time of peak drywell pressure throughout the range of power/flow operating conditions with the extended power uprate. A more detailed computer model of the NSSS (LAMB or TRACG) may be used to determine more realistic RPV break flow rates for input to the M3CPT code. The use of LAMB code has been reviewed by the NRC for application to LOCA analysis in accordance with 10 CFR Part 50, Appendix K. The results from these analyses will also be used for input to the LOCA dynamic loads evaluation.

Appendix G of NEDC-32424 also requires the applicant to perform long-term containment heatup (suppression pool temperature) analyses for the limiting FSAR events to show that pool temperatures will remain within limits for containment design temperature, ECCS NPSH and equipment qualification temperatures. These analyses can be performed using the GE computer code SHEX. The SHEX computer code for the calculation of suppression pool response to LOCA events can be approved on a plant-specific basis, provided that confirmatory calculations for validation of the results were included in the plant-specific request. To obtain approval of the use of SHEX, the licensee performed a benchmark analysis as described herein. 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 analyses of containment system performance are presented below.

2.9.1.1 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 Hatch FSARs. The short-term analysis was 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 (DBA LOCA), while the long-term analysis was performed to determine the peak pool temperature response considering decay heat addition.

The licensee indicated that the analyses were performed in accordance with Regulatory Guide 1.49 and NEDC-32424 using GE codes and models. The M3CPT code was used to model the short-term containment pressure and temperature response. The more detailed RPV model (LAMB) was used for determining the vessel break flow for input to the M3CPT code in the containment analyses. The use of the LAMB model is justified in "General Electric Company Analytical Model for Loss-of Coolant Accident Analysis in Accordance with10CFR50 Appendix K," NEDE-20566-P-A, September 1986. The staff finds the use of the LAMB model detailed RPV break flow input to the M3CPT code in the containment analysis for the extended power uprate to be acceptable.

The licensee also indicated that the SHEX code was used to model the long-term containment pressure and temperature response. The original FSAR analyses used a predecessor to the HXSIZ code. However, the predecessor code is no longer available for performing confirmatory calculations. To validate the use of SHEX for Hatch containment analyses, a benchmark analysis was performed with input assumptions and values consistent with the input used in the accepted original FSAR analysis for Hatch Unit 2. Hatch Unit 2 FSAR was chosen because it is quite similar to Unit 1 and contained more detailed information than the Unit 1 FSAR with regard to the geometry inputs, power level, and system operation. Inputs to the benchmark analysis were taken from the extended power uprate analysis if the FSAR did not contain these values. Benchmark analyses were performed using the May-Witt decay heat model. The analyses predicted the peak suppression pool temperature to be within 1 °F (210.3 °F with SHEX code compared to 209.8 °F with the original FSAR analysis). The benchmark analysis also predicted the maximum long-term wetwell pressure by SHEX, using FSAR input assumptions, within 0.5 psi from the FSAR value at the time of maximum pool temperature (12.4 psi versus 12.0 psi). The containment pressure response from the onset of sprays (600 sec) to 10,000 sec was 1 to 3 psi higher than the values reported in the FSAR. This was mainly due to the assumption used in the FSAR analysis that the drywell and wetwell temperatures are instantaneously equal to the spray temperature from the onset of sprays. SHEX mechanistically models spray heat transfer, and thus, the results reflect the time required to bring the airspace temperature to spray temperature. The shape of the long-term suppression pool temperature curve and the shape of the pressure curve from the SHEX benchmark analyses matches well with the corresponding curves reported in the FSAR. Based on the preceding review, the staff finds the use of the SHEX code for the Hatch extended power uprate acceptable.

2.9.1.2 Long-Term Suppression Pool Temperature Response

(1) Bulk Pool Temperature

The licensee indicated that the long-term bulk suppression pool temperature response was evaluated for the DBA LOCA. A bounding analysis was performed at 102 percent of 2763 MWt using the SHEX code and the more realistic decay heat model (ANS/ANSI 5.1). The initial submittal analysis did not include an uncertainty adder of 2σ to the decay heat model. The staff considers the use of ANS 5.1 without an adder to represent a best estimate of the decay power. This was deemed to be inappropriate. To be consistent with the conservative nature of the calculations, it has been determined than an uncertainty adder of 2σ is necessary when using the ANS 5.1-1979 nominal decay heat model for the extended power uprate. In letters dated

July 6 and July 31, 1998, the licensee updated the initial containment analysis results with a decay heat model, that now includes the 2σ uncertainty adder.

The revised long-term containment response analysis at the extended uprated power was performed separately for Unit 1 and Unit 2 using the SHEX code and ANS 5.1-1979 decay heat model with the 2σ uncertainty adder. The analysis also used the same input values as used in the previously accepted analysis. The analysis shows that at extended uprated power, Unit 1 peak pool temperature will increase to 207 °F and Unit 2 peak pool temperature will increase to 206 °F from the current peak pool temperature of 202 °F. The revised peak pool temperature for Unit 1 is 1 °F higher than Unit 2 mainly because Unit 1 has a slightly smaller pool than Unit 2. These temperatures are below the torus design temperature of 281 °F for Unit 1 and 340 °F for Unit 2.

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 for the extended power uprate.

(2) Local Suppression Pool Temperature with SRV Discharge

A local pool temperature limit for SRV discharge is specified in NUREG-0783 because of concerns resulting from unstable condensation observed at high pool temperatures in plants without quenchers. Elimination of this limit for plants with quenchers on the SRV discharge lines is justified in GE report NEDO-30832, "Elimination of Limit on Local Suppression Pool Temperature for SRV Discharge with Quenchers." In a safety evaluation dated August 29, 1994, the staff eliminated the maximum local pool temperature limit for plants with quenchers on the SRV discharge lines, provided the ECCS pump suctions are below the quencher elevation. The licensee indicated that since both units of Hatch have quenchers above the ECCS pump suctions, no evaluation of this limit is necessary. Based on the above review, the staff concludes that no additional analyses are necessary and the extended power uprate analyses are acceptable relative to this issue.

2.9.1.3 Containment Gas Temperature Response

The licensee indicated that the containment gas temperature response analyses were performed to cover the blowdown period for DBA-LOCA during which the maximum drywell airspace temperature occurs at 102 percent of extended uprate power using the Mark I containment LTP methodology. The results show that for Unit 1, the calculated peak drywell gas temperature at extended uprated power increases by 1 °F, from 292 °F to 293 °F, from the current power level. For Unit 1, the calculated peak drywell gas temperature exceeds the drywell shell design temperature of 281 °F by 12 °F, but only at the beginning of the accident and for a short period of time. For the worst case, the exceedance is less than 20 seconds. The drywell shell is an extremely massive steel structure about 1.5 inches thick. As a result, the shell temperature will lag the gas temperature. This thermal lag will cause the shell to not see the exceedance of such a small change in temperature for such a small time period. Thus, the shell design temperature will not be exceeded. Calculations of the shell temperature transient for a steamline break are also provided below. For this more severe case, it was shown that the design shell temperature of 281 °F was not exceeded. For Unit 2, the calculated peak drywell temperature at extended uprated power remains at 292 °F and below the drywell shell design temperature of 340 °F.

The licensee also indicated that drywell temperature responses were calculated for a series of small steamline breaks, which produced higher drywell gas temperature than the DBA LOCA. The peak drywell gas temperature of 324 °F was calculated at 102 percent of the extended uprate power level with containment parameters that bound Units 1 and 2. This temperature is below the drywell shell temperature of 340 °F for Unit 2, but exceeds the drywell shell design temperature of 281 °F for Unit 1 for a short duration. Therefore, an analysis was performed to determine the shell temperature response. The calculated peak drywell shell temperature determined for the corresponding peak drywell gas temperature of 324 °F was 271°F. This temperature is within the drywell shell design temperature of 281 °F for Unit 1. Therefore, the drywell gas temperature response for the extended power uprate has no adverse effect on the containment structure.

The licensee stated that the wetwell gas space peak temperature was calculated assuming thermal equilibrium between the pool and the wetwell gas space. The extended uprate analysis shows that the bulk suppression pool and space temperature will increase slightly after a LOCA to 207 °F for Unit 1 and 206 °F for Unit 2 and would remain below the design value of 281 °F for Unit 1 and 340 °F for Unit 2. Therefore, the wetwell gas temperature responses at the higher power level have no adverse effect and remain acceptable.

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

2.9.1.4 Short-Term Containment Pressure Response

The licensee indicated that the short-term containment response analyses were performed for the limiting DBA LOCA. These analyses assume a double-ended guillotine break of a recirculation suction line to demonstrate that operation at the proposed power level will not result in exceeding the containment design limits. The short-term analyses cover the blowdown period during which the maximum drywell pressure and maximum differential pressure between the drywell and wetwell occur. These analyses were performed at 102 percent of the extended power uprate level using previously accepted methods. These methods were originally accepted during the Mark I containment long-term program (LTP). Break flow was calculated using a more detailed RPV model . The extended power uprate analyses calculated a maximum containment pressure of 50.5 psig for Unit 1, as proposed in TS 5.5.12, and 46.9 psig for Unit 2, as proposed in TS 5.5.12, from the current values of 49.6 psig for Unit 1 and 45.5 psig for Unit 2. These calculated maximum uprate pressures for Unit 1 and Unit 2 remain below the containment design pressure of 56 psig for both units.

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

2.9.2 Containment Dynamic Loads

2.9.2.1 LOCA Containment Dynamic Loads

Generic guidelines in NEDC-32424 specify that the extended power uprate applicant determine if the containment pressure, suppression pool temperature, and vent flow conditions calculated with the M3CPT code for the extended 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 extended power uprate and, thus, do not require further analysis.

LOCA containment dynamic loads for the extended 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 dynamic loads considered in the extended power uprate evaluations include pool swell, condensation oscillation (CO), and chugging. For a Mark I plant like Hatch, vent thrust loads are also evaluated.

The licensee stated that the short-term containment response conditions with extended power uprate are within the range of test conditions used to define the pool swell and CO loads for the plant. The long-term response conditions with extended power uprate in which chugging would occur are within the conditions used to define the chugging loads. The vent thrust loads for extended power uprate are calculated to be less than the plant-specific values determined during the Mark I Containment LTP. Therefore, the LOCA dynamic loads for Hatch are not impacted by the extended power uprate.

Based on the preceding review, the staff concludes that the LOCA containment dynamic loads will remain acceptable for the extended power uprate.

2.9.2.2 SRV Containment Dynamic Loads

The SRV containment dynamic loads include 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. None of these parameters, including SRV opening setpoint, are changed for the extended power uprate for the first SRV actuation for Hatch. Therefore, the extended power uprate will not impact the SRV load definition for first actuations.

For subsequent actuations (second pops), the only additional parametric change with the extended power uprate is the time between SRV actuations. A higher water level at the time of the pop will result in higher SRV loads. The licensee stated that the effect of the extended power uprate on the SRV discharge line was evaluated. The evaluation showed that with the current SRV low-low setpoint logic parameters, the SRVDL water level will reestablish equilibrium height well before the subsequent SRV actuations. Therefore, there will be no impact of extended power uprate on the SRV subsequent actuation loads.

Based on the preceding discussion, the staff concludes that the SRV containment dynamic loads will remain acceptable for the extended power uprate.

2.9.2.3 Subcompartment Pressurization

Generic guidelines in NEDC-32424 require that the break flow will be compared with the analytical or experimental basis for the LOCA subcompartment pressurization dynamic loads. If the calculated break flow conditions with the extended power uprate are within the range of break flow conditions used to define the loads, subcompartment pressurization dynamic loads are not affected by the power uprate.

The licensee indicated that due to changes in operating conditions with the extended power uprate, the actual asymmetrical loads on the vessel, attached piping, and biological shield wall (from a postulated pipe break in the annulus between the reactor vessel and biological shield wall) will increase slightly. The biological shield wall and component designs remain adequate because the original analyzed loads were based on conservative assumptions that provide sufficient margin to accommodate the mass and energy releases at the extended power uprate conditions. Based on the preceding review, the staff concludes that the subcompartment pressurization effects will remain acceptable for the extended power uprate.

2.9.3 Containment Isolation

The licensee indicated that the system designs for containment isolation are not affected by the extended power uprate. The capability of the actuation devices to perform at the extended power uprate conditions has been evaluated and determined to be acceptable. All motor-operated valves (MOVs) used as containment isolation valves will be capable of performing their intended functions at extended uprate conditions as per the requirements of Generic Letter 89-10. The extended power uprate has no adverse effect on containment isolation. Based on its review, the staff finds that the operation of the plant at the extended uprated power level will not impact the containment isolation system.

2.9.4 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). Post-accident hydrogen and oxygen generation rates will increase in proportion to the power level. The function of Combustible Gas Control is that the system be used following a LOCA to maintain the containment atmosphere as a noncombustible mixture.

The control of combustible gas concentrations for Unit 1 is attained by the containment atmosphere dilution (CAD) method. This method adds nitrogen to the containment to dilute the oxygen concentration below the flammability limit. The licensee indicated that sufficient capacity exists in the Unit 1 CAD system to account for the increase in oxygen generation due to the extended power uprate. Under the most conservative assumptions, the extended power uprate may require the Unit 1 CAD system to be initiated earlier in the accident. The CAD system will be started when the oxygen concentration reaches 4 vol percent in about 1.55 days inside the torus and in about 2.14 days inside the drywell. These slightly reduced starting times still provide more than adequate time for the operators to take the necessary actions to initiate the CAD system. Oxygen concentration for Unit 1 is required to be controlled within 5 vol percent following a LOCA. Margin has been provided by designing the CAD system to control oxygen within 4 vol percent.

The Unit 2 combustible gas control system is provided with hydrogen recombiners which maintain a safe level of hydrogen inside the containment. The initiation of the recombiners is controlled procedurally to maintain gas concentration within 4 vol percent inside containment following a LOCA, and not by time. The licensee indicated that the impact of the extended power uprate would cause the Unit 2 recombiners to be initiated in about 1 hour 50 minutes, which is slightly earlier than the current time. Margin has been provided by designing the recombiner system to control hydrogen within 3.5 vol percent. Containment purge capability serves as backup to the Unit 1 CAD system and the Unit 2 recombiner system.

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

2.10 Engineered Safety Features and Associated Support Systems and Auxiliary Systems

2.10.1 Standby Gas Treatment System

The standby gas treatment system (SGTS) is designed to minimize offsite and control room dose rates during venting and purging of both the primary and secondary containment atmosphere under accident or abnormal conditions while containing airborne particulates and halogens that might be present. The capacity of the SGTS was selected to maintain the secondary containment at a slight negative pressure. The licensee stated that the capability of the SGTS and the charcoal filter beds would not be impacted by the extended power uprate conditions. Also, the licensee stated that the total post-LOCA iodine loading at extended power uprate conditions remains within the current capability of the filter and in accordance with Regulatory Guide 1.52, Revision 2.

Based on its review, the staff finds that plant operation at the extended power uprate level has an insignificant impact on the ability of the SGTS to meet its design objectives, and concludes that it is acceptable.

2.10.2 Fuel Pool Cooling

The spent fuel pool cooling system (SFPCS) 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. Supplemental fuel pool cooling is provided by the RHR system in the event of a full core off-load.

As a result of plant operation at the extended power uprate level, the SFP heat load would increase since the fuel discharged at each refueling outage would contain slightly more decay heat. The licensee determined that the expected heat load in the SFP would increase by approximately 8 percent for the normal condition, 9 percent for the refueling condition, and

7 percent for the maximum (core off-load) condition. Based on these increased heat loads, it was determined that the SFPSC and the RHR system in its fuel pool cooling assist mode would be able to maintain the SFP temperature within the maximum acceptable temperature of 150 °F for the normal, refueling, and maximum (core off-load) conditions. The licensee concluded that the extended power uprate would not have any negative effect on the capability to keep the SFP temperature at or below the design temperature to maintain adequate SFP cooling. Also, the licensee installed a decay heat removal system that is capable of removing 40 MBtu/hr while keeping the SFP $\leq$ 125 °F. This system would be able to adequately cool the SFP during a refueling outage for the extended power uprate operation.

Based on the results of the licensee's analysis, the staff concludes that the slight increases in the SFP heat load and fuel pool temperatures due to the extended power uprate would be within the design limits of the SFPCS, and are acceptable.

2.10.3 Water Systems

2.10.3.1 Service Water Systems

2.10.3.1.1 Plant Service Water System

The plant service water (PSW) system is designed to provide cooling water to various systems (both safety-related and nonsafety-related) and to provide makeup to the plant circulating water system. There are two divisions of PSW. Each of the two divisions supplies cooling water to one redundant train of safety-related equipment. The two safety-related headers pass through isolation valves and eventually merge into one header supplying nonsafety-related equipment. The licensee evaluated the PSW system and found that the increase in heat loads for the components affected by the extended power uprate are within the existing design heat loads, except for the stator water coolers, the isophase bus duct coolers, and the reactor building closed-cooling water (RBCCW) heat exchangers. An evaluation of the PSW system capabilities revealed that the design of the PSW system can accommodate the increase in component heat loads for these three coolers and is adequate for extended power uprate conditions.

Based on its review, the staff finds that plant operation at the extended power uprate level has an insignificant effect on the design function of the PSW systems, and concludes that it is acceptable.

2.10.3.1.2 Safety-Related Loads

The safety-related portion of the PSW system piping provides a reliable supply of cooling water to various safety-related equipment. The licensee stated that the diesel generator loads and cooling loads remain virtually the same as for current rated operation since the equipment and system performance remains unchanged. Also, PSW flow to these components would not change. The building cooling heat loads remain the same as for current rated operation because the equipment performance in these areas has remained unchanged for post-LOCA conditions. The staff finds that the safety-related loads for the PSW are not affected by the extended power uprate and the analysis is acceptable.

2.10.3.1.3 Nonsafety-Related Loads

The PSW discharge temperature results from the heat rejected via the RBCCW system and other auxiliary heat loads. The PSW heat load increase from the extended power uprate would reflect an increase in main generator losses rejected to the stator water coolers, hydrogen coolers, and exciter coolers. There would also be an increase in heat loads for the Unit 1 and 2 isophase bus duct coolers and the Unit 1 and 2 RBCCW heat exchangers. The licensee determined that the slight increase in discharge temperature between rated and uprated power in the PSW system demonstrated that the PSW system is adequate for extended power uprate conditions.

Since nonsafety-related loads do not perform any safety function, the staff has not reviewed the impact of the extended power uprate on the designs and performances of these systems.

2.10.3.1.4 RHR Service Water System

The RHR service water system is designed to provide a reliable supply of cooling water to the RHR system under normal and post-accident conditions. The licensee stated that the post-LOCA RHR cooling capacity does not increase for the extended power uprate. Therefore, the extended power uprate would not increase the cooling requirements on the RHR and its associated service water system significantly. During shutdown cooling with the RHR system, the greater decay heat generation may require a longer time to cool the reactor following extended power uprate conditions, although the impact is expected to be minimal.

Based on its review, the staff finds that the design cooling capacity of the RHR service water system is adequate for extended power uprate conditions, and concludes that it is acceptable.

2.10.4 Main Condenser/Circulating Water/Normal Heat Sink Performance

The main condenser, circulating, and cooling tower 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 load thereby maintaining low condenser pressure. The licensee stated that the performance of the main condenser, circulating water, and cooling tower systems was evaluated and found to be adequate for plant operations at the extended power uprate level.

Since the main condenser, circulating water system and cooling tower systems do not perform any safety-related function, the staff has not reviewed the impact of the extended power uprate on the designs and performances of these systems, except for environmental considerations.

2.10.5 Reactor Building Closed Cooling Water System

The RBCCW system is designed to remove heat from various auxiliary plant equipment housed in the reactor building. The licensee performed evaluations and determined that the increase in heat loads to this system due to the extended power uprate would be within the cooling capabilities and RBCCW flow rates.

Since plant operations at the proposed extended power uprate level do not change the design aspects and operations of the RBCCW system, the staff finds that plant operations at the extended power uprate level is acceptable.

2.10.6 Heating, Ventilation and Air Conditioning (HVAC)

The HVAC systems consist mainly of cooling supply, exhaust and recirculation units in the turbine building, reactor building, and the drywell. The actual heat loads processed by the HVAC systems in these buildings would be impacted by the extended power uprate due to the increased heat losses from the isophase bus ducts and increased horsepower requirements of the condensate pump, condensate booster pump and the Unit 1 reactor recirculation pump motors. The licensee's review indicated that the actual design capacities of the cooling units in the condenser bay area near the condensate and condensate booster pumps have sufficient margin to accommodate the additional heat loads due to the increased horsepower requirements without any impact on the area temperatures. Also, the increased heat loads due to an increase in isophase bus duct temperature would be within the design capacity of the coolers in the area.

Due to the increased horsepower requirements of the Unit 1 reactor recirculation pump motors, the localized bulk average temperature in the vicinity of the recirculation pumps would increase by approximately 2 °F. The licensee reviewed the actual design capacity of the drywell coolers and found that the additional Unit 1 heat load would be within the design capacity of the drywell cooling system. There would not be an increase in the HVAC load for the Unit 2 reactor recirculation pump motor windings since they are water-cooled. Other areas are unaffected by the proposed extended power uprate since the process temperatures remain relatively constant.

Based on its review, the staff finds that plant operation at the extended power uprate level has an insignificant impact on the ability of the HVAC to meet its design objectives, and concludes that it is acceptable.

2.10.7 Fire Protection

The licensee reviewed the Hatch 10 CFR Part 50, Appendix R Fire Hazard Analysis Report and the impact of the Safe Shutdown Analysis Report on Appendix R and found that operation at the extended power uprate level does not affect the ability of the Appendix R systems to perform their safe shutdown function. Fire suppression systems, fire detection systems, and operator actions required to mitigate the consequences of a fire would not be affected. There are no physical plant configuration or combustible load changes as a result of the increase in power. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change and remain adequate for the extended power uprate conditions.

In the evaluation of Appendix R events, the licensee determined that the time available for operator action to initiate containment cooling was reduced. The containment cooling was assumed to be initiated 3 hours after the event occurs instead of 4 hours as assumed in the previous Hatch Appendix R analyses. The licensee stated that the revised initiation time remains well within the expected operator response time.

Based on its review, the staff finds that the fire suppression and detection systems and their associated analyses are insignificantly impacted by the extended power uprate, and concludes that the fire protection systems are acceptable at the extended power uprate level.

2.10.8 Additional Systems Not Impacted By Extended Power Uprate

In Section 6.8 of Enclosure 6 to Reference 1, the licensee identified and evaluated plant systems that are not affected, or are affected in a minor way, by operation of the plant at the extended uprated power level. Based on its review, the staff agrees that plant operation at the proposed extended power uprate level has no impact, or an insignificant impact, on these systems.

2.10.9 Power Conversion Systems

The power conversion systems were originally designed to utilize the energy available from the nuclear steam supply system and were designed to accept the system and equipment flows resulting from continuous operation at $\geq$ 105 percent of rated steam flow. The original power uprate to 2558 MWt involved some turbine modifications. For the extended power uprate, additional turbine modifications are required to accommodate the higher steam flow.

2.10.9.1 Turbine-Generator

In 1996, the Hatch Unit 1 turbine-generator was uprated to 105 percent of its original rating. In order to achieve the extended power uprate, which is 113 percent of the original rating, the high pressure turbine steam path components require modifications of the first, second, and third-stages to increase the steam flow capacity. The moisture separator reheaters (MSRs) also require modification, which may include reheater tube bundle replacement and new moisture separator chevrons. The licensee plans to perform these modifications to Unit 1 in the spring 1999 outage. The licensee stated that the high pressure turbine and MSRs should allow Unit 1 to operate at or near the proposed power level with adequate turbine pressure control.

Hatch Unit 2 was uprated to 105 percent of its original rating in 1995. In 1997, new steam path hardware was installed in the Unit 2 turbine-generator to achieve the flow capacity corresponding to about 113 percent of its original rating. The new hardware consisted of first-stage cold assembled buckets (two rows), new second, third, and fourth-stage diaphragms, and new redesigned second and third-stage buckets. The licensee plans to modify the Unit 2 MSRs during the fall 1998 outage. The licensee stated that the modifications to the high pressure turbine and planned MSR changes should allow Unit 2 to operate at or near the proposed power level with adequate turbine pressure control.

The licensee evaluated the Hatch Unit 1 and 2 turbine-generators to assess the impact of increased steam flow and pressure as a result of the extended power uprate. The evaluations were based on the turbine control valves in the valve wide open (VWO) position for the increased power conditions. The VWO position corresponds to slightly above 108 percent of the current core thermal power of 2558 MWt. The licensee performed an overspeed calculation that evaluated the entrapped steam energy contained within the turbine and the associated piping after the stop valves trip and the sensitivity of the rotor train for the capability of overspeeding. It was determined that for the slight increases in the entrapped energy, an adjustment of the overspeed trip setting was not needed. Therefore, the licensee concluded that for the revised steam specifications for the extended power uprate, no changes to the current mechanical trip settings would be required.

Based on its review, the staff finds that the modifications to the turbine generators and the current mechanical trip settings should allow Units 1 and 2 to operate at or near the proposed power level. Therefore, the staff concludes that operation of the modified turbine-generator at the extended power uprate level is acceptable.

2.10.9.2 Miscellaneous Power Conversion Systems

The licensee evaluated the miscellaneous steam and power conversion systems and their associated components at the extended uprated power level. The systems include the condenser and steam jet air ejectors, the turbine steam bypass, and the feedwater and condensate systems. The licensee stated that the existing equipment for these systems, with minimal modifications, are acceptable for operation at the extended power uprate level.

Since these systems do not perform any safety function, the staff has not reviewed the impact of plant operations at the extended power uprate level on the designs and performances of these systems.

2.11 High Energy Line Break Outside Containment

The licensee evaluated the high energy line breaks (HELBs) against the criteria set forth in the FSARs for HELBs outside containment. The critical parameter affecting the HELB analysis for the extended power uprate is an increase in reactor vessel dome pressure. Since there would not be an increase in the analyzed dome pressure, the licensee determined that there would not be an increase in the mass and energy release rates following HELBs. The existing mass and energy release rates for HELBs outside primary containment were based on saturated fluid conditions and frictionless critical mass fluxes determined at the local reactor vessel pressure, which assumed a 1060 pounds per square inch absolute (psia) steam dome pressure. Additional conservatism was incorporated in the existing analyses by neglecting all piping losses and by assuming continuous blowdown at the maximum rate. Since a steam dome pressure of 1050 psia was used for the proposed extended power uprate, the licensee stated that the existing HELB analyses were bounding for the proposed extended power uprate conditions for the main steam system line break, the high pressure ECCS line break, the reactor core isolation cooling system line break, the reactor water cleanup system line break, pipe whips, jet impingements and the temperature, pressure, and humidity profiles.

For flooding considerations, the feedwater system line break would be the most critical case. The flooding rate would depend on the hardware in the feedwater system, such as pipes and pumps. The licensee stated that since the feedwater system hardware would not be changing substantially, the existing feedwater break flooding analysis would be valid for the extended power uprate conditions.

Based on its review, the staff finds that the mass and energy releases considered for the current power level of 2558 MWt adequately represent the releases for the extended power uprate, and concludes that it is acceptable.

2.12 Integrity of Reactor Vessel and Internals and Reactor Coolant System Piping

In Section 3.3 of NEDC-32749 and in the response to the staff's RAI, SNC assessed the effects of the Hatch Unit 1 and Unit 2 extended power uprates on each RPV and each set of reactor internals. Regarding the RPV, the licensee provided an assessment of: (1) the impact of the extended uprate on the adjusted reference temperature (ART) of the limiting RPV materials; (2) the need to revise the Hatch Unit 1 and Unit 2 pressure-temperature (P-T) limit curves; (3) the changes in the predicted upper shelf energy (USE) drop for the RPV materials and the validity of previously approved equivalent margin analyses; and (4) whether changes in the RPV surveillance program (as required by 10 CFR Part 50, Appendix H) are necessary. Regarding the reactor internals, the licensee provided an assessment of: (1) changes in pressure differential loadings caused by the extended power uprate; (2) changes in the assessment of flow-induced vibration caused by the extended power uprate; and (3) changes in the potential for erosion due to the extended power uprate.

In analyzing the RPV, SNC examined the effect on the RPV fluence of operating Hatch Unit 1 and Unit 2 at a power of 2763 MWt until end-of-license (EOL). SNC's analysis, therefore, addressed the expected RPV material embrittlement since it is directly related to the RPV neutron fluence, which is in turn related to the reactor operating power. In its March 9, 1998, letter, SNC provided the information contained in Tables 1 and 2 on the projected EOL neutron fluence at the clad-to-base metal interface ("surface fluence"), the 1/4T location (i.e., at a point one-quarter of the way through the RPV wall from the RPV inside diameter) and the 3/4T location (i.e., at a point three-quarters of the way through the RPV wall from the RPV inside diameter) for each Hatch unit.

In its August 8, 1997, submittal, SNC concluded that "[A] comprehensive review...[of] the reactor vessel and internals...show continued compliance with the original design and licensing criteria for the reactor vessel." SNC went on to explain that with regard to the application of the requirements of Title 10 of the <u>Code of Federal Regulations</u> Part 50, Appendix G, "Fracture Toughness Requirements" (10 CFR 50, Appendix G) to the Hatch Unit 1 and Unit 2 RPV materials:

(a) The BWR Owner's Group Equivalent Margin Analyses are applicable and demonstrate that the upper shelf energy (USE) maintains the margin requirements in 10 CFR 50 Appendix G for the design life of the vessel.

(b) The 32 effective full power years (EFPY) shift is slightly increased and, consequently requires a change in the adjusted reference temperature (ART), which is the initial RT_{NDT} plus the shift [NRC note: plus the margin term]. The beltline material adjusted reference temperature (ART) will remain well within the

200 °F regulatory requirement [NRC note: This is no longer included in 10 CFR [Part] 50 as a regulatory requirement].

(c) The current P-T curves have been revised considering the slight increase in shift affecting the beltline portion of the curves. This increase in beltline shift affected the P-T curves for Unit 1 beyond 20 EFPY to the end of life and the P-T curves for Unit 2 beyond 28 EFPY to the end of life.

To address these points, SNC submitted an analysis intended to bound the USE drops for the limiting base and weld metal for each Hatch unit. For this evaluation, SNC chose to conservatively assess the limiting base and weld metal (the ones with the highest copper content) for each unit by assigning them the peak vessel clad-to-base metal fluence. This was considered to be conservative since the evaluation of USE drop for each material would permit SNC to use the 1/4T fluence for each material in its evaluations. Based on this approach, SNC demonstrated that the predicted percentage USE drops did not exceed the amount permitted under General Electric Topical Report, NEDO-32205-A, "10CFR50 Appendix G, Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," as it was approved by the NRC staff. For the pressure-temperature limits, SNC submitted new pressure-temperature limits for heatup/cooldown, core criticality, and hydrostatic testing at Hatch Unit 1 for 20, 24, 28, and 32 EFPY. Similar curves were submitted for Unit 2 for 32 EFPY only.

Finally, in its March 9, 1998, letter, SNC stated that when considering the possibility of a change to the RPV surveillance capsule withdrawal schedule for Unit 2 (the only unit to have a capsule yet to be pulled before EOL), "...it will not be necessary to change the removal interval due to the increased fluences associated with extended power uprate conditions."

For the RPV internals, SNC stated that the proposed power uprate will have no adverse effect on the flaw evaluations for existing flaws in the core shroud or in the core spray header. SNC also concluded that the extended uprate would not affect the operation of any other reactor internals. SNC's evaluation of the reactor coolant system piping confirmed that changes in the flow parameters associated with the extended power uprate would have no significant effects on the potential for flow-induced erosion/corrosion in those systems that might be susceptible to the phenomenon (e.g., the feedwater or main steam system).

The staff has reviewed the RPV assessment provided by SNC and determined that the changes to the P-T limit curves and the changes to the upper shelf energy analyses for each unit are acceptable and that the RPV assessment addressed the changes in RPV material embrittlement due to the extended power uprate conditions. The staff has also concluded that the extended power uprate does not necessitate any change in the Hatch Unit 1 or Unit 2 10 CFR Part 50, Appendix H, RPV surveillance program.

In accordance with 10 CFR Part 50, Appendix G, RPV materials must maintain a USE of 50 ft-lbs throughout the life of the vessel or demonstrate that lower values provide margins of safety equivalent to those required by Appendix G to Section XI of the ASME Code. Based upon the 2763 MWt 1/4T fluence values submitted by SNC, the staff concluded that the (percentage) upper shelf energy drop for the limiting Hatch Unit 1 RPV weld and base metal materials at EOL submitted by SNC was consistent with or more conservative than the results achieved by the staff using the Regulatory Guide 1.99, Revision 2 methodology. However, in the analysis for the limiting weld of Hatch Unit 1, there was a significant difference between SNC's and the staff's evaluations. SNC used a copper weight percent value of 0.28 for the limiting heat 1P2815 lower-intermediate shell axial weld and the peak vessel clad-to-base metal fluence of 1.94 x 10^{18} n/cm^2 (at 32 EFPY) to arrive at a USE drop of 29 percent. The staff, based on its examination of the Combustion Engineering Owners Group report, CEOG NPSD-1039, Revision 2, "Best-Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," noted that a best-estimate copper value for this weld wire heat based on all available information was given as 0.316 weight percent. Using the 0.316 weight percent copper value and the 1/4T fluence for the 1P2815 weld at EOL of 1.4 x 10^{18} n/cm^2, the staff also concluded that the USE drop would be 29 percent. By either methodology, this was less than the allowable drop of 34 percent documented in NEDO-32205-A as acceptable for BWR/4 RPV weld materials. The NRC staff, as part of a separate action to resolve issues related to Generic Letter 92-01, "Reactor Vessel Integrity," has issued RAIs to SNC and other licensees that address issues regarding best-estimate chemistries. Since the specific chemistry for the 1P2815 weld does not adversely affect its USE evaluation when the other conservatism in the SNC approach is taken into account, the staff has chosen to not pursue resolution of the weld chemistry issue in the process of the extended power uprate review, but will revisit the issue in the Generic Letter 92-01 RAIs. Therefore, SNC has demonstrated that the Hatch Unit 1 and Unit 2 RPV materials retain margins of safety equivalent to those required by ASME Code, Section XI, Appendix G.

In evaluating the effect of the extended power uprate on the shift in limiting materials adjusted reference temperature and the need for new P-T limit curves, the staff applied the methodology found in Regulatory Guide 1.99, Revision 2, for evaluation of radiation embrittlement. This methodology specifies that a material's ART can be determined from the following equation:

$$\text{ART} = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

where,

$$\Delta RT_{NDT} = CF \times FF$$

with, CF = the Chemistry Factor, a function of copper and nickel content
FF = the Fluence Factor, a function of the RPV material's neutron fluence

and, Margin = a value added to account for analytical uncertainties.

For Hatch Unit 1, the limiting material was found to be plate G-4803-7. This plate has an assigned CF of 334 °F based upon the application of the credible Hatch Unit 1 plate surveillance data to its CF determination. The methodology used by SNC to establish this value was reviewed and approved in a staff SE dated August 19, 1997, and since the extended power uprate conditions do not affect the CF determination, it was not rereviewed in this analysis. However, the staff has asked for additional information regarding the licensee's methodology for establishing these RPV capsule fluences and will revisit this issue if significant concerns are discovered. Given the margins and factors of safety, which are explicitly incorporated into the determination of P-T limits curves (K_{IR} fracture toughness curve, factors of 1.5 to 2 on pressure stresses, 1/4T flaw postulation, etc.) and the conservatism in the licensee's analysis, the staff has determined that the application of a CF of 334 °F is sufficient pending additional staff review. Note, the difference in chemistry assigned the 1P2815 weld by the licensee and the NRC (as mentioned in the USE analysis) did not affect the P-T limits evaluation since Plate G-4803-7 continued to be the limiting material even when the higher copper content was evaluated by the NRC staff.

Also, given that the credible surveillance data was still being used, the margin term assigned in the analysis did not change from that in the August 19, 1997, SE. However, the fluences (and fluence factors) associated with the beltline materials at 20, 24, 28, and 32 EFPY of operation did change due to the extended power uprate. Based upon the Unit 1 fluences provided in the March 9, 1998, letter, the staff checked the heatup/cooldown, core criticality, and hydrostatic test curves provided by SNC in Enclosure 4 to the original extended power uprate submittal (Reference 1). The staff found that the curves submitted by the licensee were consistent with, or conservative to, those determined by the staff using the methodology of the 1989 Edition of the ASME Code, Section XI, Appendix G. Those portions of the curve, which were determined by consideration of the non-beltline regions of the RPV (bottom head, flange, etc.), were checked against those previously approved by the staff and also found to be acceptable. The curves approved by the staff for Unit 1, as proposed in TS 3.4.9, have been included as Figures 1, 2, and 3.

For Unit 2, the CFs for all materials were determined by using the Table CFs in Regulatory Guide 1.99, Revision 2, since no credible surveillance data exists for either weld or plate materials. In the staff's most recent SE on updating the Hatch Unit 2 P-T curves (dated April 4, 1997), the staff noted that the limiting material for the Hatch Unit 2 RPV was lower shell course axial weld 101-842 fabricated using weld wire heat number 10137. In that analysis, SNC had assigned the peak RPV 1/4T fluence value to all of the beltline axial welds. In the extended power uprate submittal, SNC differentiated between the peak fluences for the axial welds in the lower-intermediate shell course (0.157×10^{19} n/cm^2 at the 1/4T location, ff=0.513) and the axial welds in the lower shell course (0.0947×10^{19} n/cm^2 at the 1/4T location, ff=0.406). Because of this differentiation, the lower-intermediate shell course axial welds manufactured with weld wire heat 51874 became the limiting material for the Hatch Unit 2 vessel. SNC reported a chemistry for the 10137 weld of 0.23 wt percent Cu, 0.50 wt percent Ni, CF=155 and a chemistry for the 51874 weld of 0.18 wt percent Cu, 0.50 wt percent Ni, CF=138.

Based upon the Unit 2 fluences provided in the March 9, 1998, SNC letter, the staff checked the heatup/cooldown, core criticality, and hydrostatic test curves provided by SNC in Enclosure 4 to the original extended power uprate submittal. The staff found that the curves submitted by the licensee were consistent with or more conservative than those determined by the staff using the methodology of the 1989 Edition of the ASME Code, Section XI, Appendix G. Those portions of the curve, which were determined by consideration of the non-beltline regions of the RPV (bottom head, flange, etc.), were checked against those previously approved by the staff and also found to be acceptable. The curves approved by the staff for Unit 2, as proposed in TS 3.4.9, have been included as Figures 4, 5, and 6.

The staff accepts the application of the fluences proposed by SNC in its March 9, 1998, submittal for each Hatch unit. Given the margins and factors of safety, which are explicitly incorporated into the determination of P-T limits curves (K_{IR} fracture toughness curve, factors of 1.5 to 2 on pressure stresses, 1/4 T flaw postulation, etc.) and the conservatism in the licensee's analysis, the staff has determined that these fluences are sufficient. In addition, in the case of Unit 2, the staff has noted that the chemistry values proposed by SNC for the two weld wire heats previously noted (51874 and 10137) are both significantly higher than the best-estimate chemistries reported in the 1997 Combustion Engineering Owners Group report NPSD-1039, Revision 2, "Best-Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds." This may provide additional conservatism in the case of the curves proposed for Unit 2. An RAI has been issued to Hatch (April 24, 1998) as a follow-up to Generic Letter 92-01, Revision 1, Supplement 1 to resolve these chemistry issues.

Finally, based on the information cited above regarding the change in RPV fluence, the staff finds that no modification of the Hatch Unit 1 or Unit 2 RPV surveillance program is necessary due to the extended power uprate. However, the staff would note that some concerns have been raised regarding BWR RPV surveillance programs in general with the Boiling Water Reactor Vessels and Internals Project (BWRVIP). These issues question whether certain BWRs possess unirradiated baseline surveillance data from which to measure changes in RPV material embrittlement and, if the data is not available, what actions can or should be taken to address the issue. Hatch Unit 1 has been identified by the staff as potentially lacking baseline data for some surveillance program materials. While this topic does not directly affect the status of the extended power uprate review, SNC is participating in the BWRVIP to address this outstanding issue.

The staff has reviewed the licensee's evaluations regarding the effect of the extended power uprate on core shroud and core spray piping and concludes that the licensee has bounded the effects of the extended power uprate on the existing flaws. The staff concludes that the proposed extended power uprate will not affect the operation of core shroud, core spray header, or any other RPV internals.

The proposed extended power uprate will slightly increase the susceptibility of piping to erosion/corrosion (E/C), but since SNC has reexamined its E/C inspection programs in light of plant-specific extended power uprate concerns (i.e., increased flow-induced E/C in systems associated with the turbine cycle) and made appropriate adjustments to these programs, the licensee has determined that there will be a negligible effect on E/C.

Based on the preceding information, the staff has concluded that RPV integrity, RPV internals, and reactor coolant system erosion/corrosion issues have been adequately addressed in the SNC submittal and that the extended power uprate is acceptable.

2.13 Structural and Pressure Boundary Integrity

The staff's review included an evaluation on the effects of the extended 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 CRD mechanism, certain pumps and valves, and the BOP piping systems.

2.13.1 Reactor Pressure Vessel and Internals

The licensee evaluated the reactor vessel and internal components in accordance with the current licensing basis. Load combinations include reactor internal pressure difference (RIPD), LOCA, and seismic loads. The seismic loads are unaffected by the extended power uprate. The licensee recalculated RIPDs for the proposed extended power uprate are shown in Tables 3-2, 3-3, and 3-4 of NEDC-32749P (Enclosure 6 to Reference 1), for normal, upset and faulted conditions, respectively.

The stresses and cumulative fatigue usage factors (CUFs) for the reactor internal and vessel components were evaluated by the licensee in accordance with the codes of record at Hatch, the ASME Code, Section III, 1965 Edition with Winter 1966 Addenda for Hatch Unit 1 and 1968 Edition with 1970 Addenda for Hatch Unit 2. The load combinations for normal, upset, and faulted conditions were considered in the evaluation. The maximum stresses for critical components of the reactor internals were summarized in Table 3-1 of NEDC-32749P and Table 19-1 of Reference 2 for the current operating and extended power uprate conditions. The calculated stresses are less than the allowable code limits shown in the table. In Reference 2, the licensee indicated that the proposed extended power uprate evaluation was performed for the current reactor configuration incorporating recent shroud repair modifications based on the original repair modification analysis that was previously approved by the staff.

In Reference 2, the licensee also indicated that the Unit 1 feedwater nozzle, control rod drive nozzle, and vessel shell, and the Unit 2 feedwater nozzle, closure vessel shell, closure region bolts, and basin seal skirt were reanalyzed for the proposed extended power uprate. For these limiting components, the licensee provided the calculated CUFs in Table 3-5 of NEDC-32749P and the calculated stresses in Tables 24-1 and 24-2 of Reference 2. The staff finds that the calculated CUFs and stresses provided by the licensee are within the Code allowable limits. In Reference 2, the licensee indicated that the feedwater nozzle evaluation uses the 1974 ASME Code edition with addenda to and including Summer of 1976 for Unit 1, and 1971 ASME Code edition with addenda to and including Summer 1973 for Unit 2. The licensee also provided a description of the methodology for evaluating the structural integrity of the reactor vessel and components for the requested extended power uprate. The staff finds that the methodology used by the licensee is consistent with the NRC-approved metnodology in Appendix I of Reference 9, and is therefore acceptable.

In Reference 2, the licensee stated that the CUF for the Unit 2 feedwater nozzle was initially calculated to be greater than 1.0 using the design basis approach, and was recalculated to be 0.93 for 40 years of operation by combining CUFs based on the actual cycle's counting record during plant operation and based on the design basis cycles for the proposed extended power uprate condition. The staff finds that the method of counting actual cycles has been used previously by Hatch and other nuclear plant facilities to compute CUF, and the CUFs so calculated are realistic and acceptable.

The licensee assessed the potential for flow-induced vibration based on the GE prototype plant vibration data for the reactor internal components recorded during startup testing and on operating experience from similar plants. The vibration levels were calculated by extrapolating the recorded vibration data to the extended power uprate conditions and compared to the plant allowable limits for acceptance. The licensee found the maximum flow induced vibration at the jet pump riser braces to be within the acceptance limit for the Hatch proposed power uprate condition.

Based on its review of the information provided by the licensee, the staff finds that the maximum stresses and fatigue usage factors are within the Code-allowable limits, and concludes that the reactor vessel and internal components will continue to maintain their structural integrity for the extended power uprate condition.

2.13.2 Control Rod Drive System

The licensee indicated that the CRD mechanisms (CRDMs) have been designed in accordance with the codes of record, the ASME Code, Section III, 1965 Edition with addenda to and including Summer 1966 for Unit 1, and 1968 Edition with addenda to and including Summer 1970 for Unit 2. The components of the CRDM, which form part of the primary pressure boundary, have been designed for a dome pressure of 1250 psig, which is higher than the reactor bottom head pressure of 1075 psi for normal and the extended uprated power conditions.

In Reference 2, the licensee indicated that the maximum calculated stress for the CRDM indicator tube is 20,790 psi, which is less than the allowable stress limit of 26,060 psi. The maximum stress on these components results from a maximum CRD internal hydraulic pressure of 1750 psig with no other event having a significant impact on the total load. The analysis of cyclic operation of the CRDM resulted in a maximum CUF of 0.15 for the limiting CRD main flange for the power uprate. This is less than the Code-allowable CUF limit of 1.0.

On the basis of its review, the staff concludes that the CRDM will continue to meet its design basis and will continue to maintain its structural and pressure integrity at the extended uprated power conditions.

2.13.3 Reactor Coolant System Piping and Components

The licensee evaluated the effects of the extended power uprate condition, including higher flow rate, temperature, pressure, fluid transients, and vibration effects on the reactor coolant pressure boundary (RCPB) and the BOP piping systems and components. The components evaluated included equipment nozzles, anchors, guides, penetrations, pumps, valves, flange connections, and pipe supports. The evaluation was performed using the original codes of record specified in the Hatch FSARs, the code allowables, and analytical techniques. No new assumptions were introduced that were not in the original analyses.

The RCPB piping systems evaluated include main steam piping, reactor recirculation piping, feedwater piping, RPV bottom head drain line, RWCU, reactor vessel head vent line, reactor core isolation cooling (RCIC), core spray piping, HPCI piping, RHR, SRV discharge piping, and CRD piping. The evaluation included appropriate components, connections, and supports. The licensee's evaluation of the RCPB piping systems consisted of comparing the increase in pressure, temperature, and flow rate against the same parameters in the original design-basis analyses. The percentage increases in pressure, temperature, and flow for affected limiting piping systems were identified in Tables 3-6 and 3-7 of Reference 1.

As summarized in Tables 3-6 and 3-7, a majority of the RCPB systems were originally designed to maximum temperatures and pressures that bound the increased operating temperature and pressure due to the extended power uprate. For those systems whose design temperature and pressure did not envelop the extended power uprate conditions, the licensee performed stress analyses in accordance with the requirements of the Code and the Code addenda of record for the extended power uprate conditions. The licensee found that the original design analyses have a sufficient margin between calculated stresses and ASME-allowable limits to justify operation at the higher operating flow, pressure, and temperature for the proposed power uprate. The licensee indicated that all safety-related piping was analyzed in accordance with the ASME Code stress and fatigue usage criteria using the process described in the GE generic evaluation, NEDC-32523, Appendix K. The licensee concluded that the evaluation showed compliance with all appropriate Code requirements for the piping systems evaluated and that the extended power uprate will not have an adverse effect on the reactor coolant system piping design. The staff reviewed selected portions of the licensee's evaluation and finds that the licensee's conclusions are acceptable.

The licensee evaluated the stress levels for BOP piping and appropriate components, connections, and supports in a manner similar to the evaluation of the RCPB piping and supports, which is based on increases in temperature and pressure from the design basis analysis input. The BOP systems evaluated include lines which are affected by the extended power uprate, but not evaluated in Section 3.5 of Reference 1, such as feedwater heater piping, main steam bypass lines, and portions of main steam, recirculation, feedwater, RCIC, HPCI, and RHR systems outside the primary containment. The limiting stress ratios of maximum calculated stresses to the allowable, resulting from the BOP piping evaluations for the extended power uprate, are shown in Table 29-1 of Reference 2. The licensee concluded that all piping is below the Code-allowable limits. The staff finds that the stress ratios provided by the licensee are within the Code-allowable limits and are, therefore, acceptable.

The licensee evaluated pipe supports such as snubbers, hangers, struts, anchorages, equipment nozzles, guides, and penetrations by evaluating the piping interface loads due to the increases in pressure, temperature, and flow for affected limiting piping systems. The licensee indicated that there is an adequate margin between the original design stresses and Code limits of the supports to accommodate the load increase and as such, all evaluated pipe supports were within the Code-allowable limits. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the extended power uprate, and no new pipe break locations were identified. The staff finds the licensee's evaluation to be acceptable.

Regarding the assessment of the main steam flow restrictor, the licensee stated that there is no impact on the structural integrity of the restrictor for the extended power uprate. In Section 3.2 of the extended power uprate license amendment request, the licensee indicated that a higher peak RPV transient pressure of 1325 psig results from operation at 2763 MWt, but this value remains below the ASME Code limit of 1375 psig. Therefore, the main steamline flow restrictor will maintain its structural integrity following the extended power uprate since the restrictor was designed for a differential pressure of 1375 psig, which envelops the evaluated extended power uprate conditions.

Based on the preceding review, the staff concludes that the design of piping, components, and their supports will be adequate to maintain the structural and pressure boundary integrity of the BOP and reactor coolant piping, components, and supports in the proposed extended power uprate.

2.13.4 Equipment Seismic and Dynamic Qualification

The licensee evaluated equipment qualification for the extended power uprate condition. The dynamic loads such as SRV discharge and LOCA loads (including pool swell, condensation oscillation, and chugging loads) that were used in the equipment design will remain unchanged as discussed in Section 4.1.2 of NEDC-32749P because the plant-specific hydrodynamic loads defined during the Mark I Containment LTP for the design-basis analysis at Hatch are bounding for the extended power uprate.

Based on its review of the proposed extended 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 extended power uprate conditions for the following reasons:

1. Seismic loads are unchanged for the extended power uprate;
2. No new pipe break locations or pipe whip and jet impingement targets are postulated as a result of the extended uprated conditions;
3. Pipe whip and jet impingement loads are within design values for the extended power uprate; and
4. SRV and LOCA dynamic loads used in the original design-basis analyses are bounding for the extended power uprate.

2.13.5 Safety-Related SRV and Power-Operated Valves

In Reference 2, the licensee indicated that the current approved SRV setpoint tolerance (+/-3 percent) was incorporated in the abnormal transient and accident analyses at the extended uprate conditions. The licensee determined that peak RPV steam pressure remains below the ASME allowable of 110 percent of design pressure and that safety-related SRV operability is not affected by the proposed changes. The licensee stated that the plant-specific analyses for the extended power uprate condition conservatively assume one SRV out of service. This additional margin in the plant-specific analyses provides reasonable assurance that the postulated SRV setpoint drift would not result in the maximum allowable system pressure being exceeded. Furthermore, the maximum reactor dome pressure remains unchanged for the Hatch extended power uprate. Consequently, the licensee concluded that the SRV setpoints and analytical limits are not affected by the proposed extended power uprate, and that the SRV loads for the SRV discharge line piping will remain unchanged.

Based on its review, the staff concludes that the SRVs and the SRV discharge piping will continue to maintain the structural integrity and will continue to provide sufficient overpressure protection to accommodate the proposed extended power uprate.

In Reference 2, the licensee indicated that the extended power uprate will not increase any system operating pressure, reactor pressure, or safety relief valve setpoints. The licensee also stated that the valve differential pressures and line pressures were not affected by the extended power uprate. The licensee provided a table indicating the impact of the extended power uprate on plant systems and components. Based on its review, the licensee concluded that revision of the MOV and air-operated valve (AOV) programs at Hatch were not necessary. The licensee provided follow-up information in Reference 4.

In 1995, the NRC staff evaluated the Hatch MOV program developed in response to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" during inspections in February 1995 (documented in Inspection Report 50-321,366/95-02) and December 1995 (documented in Inspection Report 50-321,366/95-25). Because the staff's inspections on the Hatch MOV program under GL 89-10 were conducted prior to full implementation of the initial power uprate, the licensee described, in Reference 4, changes to the MOV program at Hatch resulting from the increased reactor pressure associated with the initial power uprate, which was approved by the NRC on August 31, 1995 (Reference 15).

The licensee reviewed the plant-specific information on Hatch systems and components for the extended power uprate to determine its potential effect on the performance of mechanical components. In References 2 and 4, the licensee concluded that there will be no significant effect on pumps and valves at Hatch from the extended power uprate. Further, the licensee evaluated changes in environmental temperatures and MOV electric power supplies and found that the proposed extended power uprate has no impact on the licensee's GL 95-07 ("Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves," dated August 17, 1995) evaluation regarding valve pressure locking or thermal binding. The licensee also indicated that it will incorporate the proposed extended power uprate condition in the evaluation in response to GL 96-06 ("Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions," dated September 30, 1996) on potential overpressurization of isolated piping segments at Hatch. The NRC staff will complete the review of the licensee's response to GL 95-07 and GL 96-06 separate from this extended power uprate request.

Based on its review and the information provided by the licensee, the staff concludes that the proposed extended power uprate will not have an adverse effect on the performance of mechanical components of safety-related pumps and valves at Hatch.

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

2.14 Instrumentation and Control

The setpoint changes for the identified instrumentation for the new power level are predicated on the assumption that analytical limits used by the licensee are based on the application of approved design codes.

The following TS changes have been proposed by the licensee:

1. TS Section 3.3.4.1, End-of-cycle recirculation pump trip

   Applicability for Limiting Condition for Operation 3.3.4.1, Required Action C.2 and Surveillance Requirement (SR) 3.3.4.1.2, the operability requirements for end-of-cycle recirculation pump trip have been changed from 30 percent to 28 percent.

2. TS Section 3.3.1.1, RPS Instrumentation, Required Action E.1, Surveillance Requirement 3.3.1.1.11, and Functions 8 and 9 of TS Table 3.3.1.1-1

   The power level at which the direct scram, for Turbine Stop Valve closure and Turbine Control Valve fast closure, is bypassed has been reduced from 30 percent to 28 percent.

3. TS Table 3.3.1.1-1, Function 2.b, Average Power Range Monitor (APRM) Simulated Thermal Power-High

   For two-loop operation, the allowable value has been changed from 0.58w+62 percent to 0.58w+58 percent.

   For single loop operation, the allowable value has been changed in footnote (b) from 0.58w+62 percent-0.58w to 0.58w+58 percent-0.58Δw.

In addition to the preceding changes, the licensee will implement new setpoints for the instrumentation which are listed in the TS as percentage of flow or pressure. However, no changes to the TS are needed since the setpoints, listed in the TS as percent, are not being changed. The licensee has identified this instrumentation as follows:

(a) APRM Scram
(b) Main Steamline High Flow

The licensee has also revised the associated TS Bases to be consistent with the changes to the TS. Reference 1 identified that the instrument setpoint methodology is the same as that used for the original 5 percent power uprate. The staff has previously reviewed this instrument setpoint methodology and found it acceptable for establishing new setpoints in power uprate applications.

The proposed setpoint changes resulting from the extended power uprate are intended to maintain existing margins between operating conditions and the reactor trip setpoints and do not significantly increase the likelihood of a false trip nor failure to trip upon demand. Therefore, the existing licensing basis is not affected by the setpoint changes to accommodate the extended power uprate and the changes are acceptable.

Based on the preceding review and justifications, the staff concludes that the licensee's instrument setpoint methodology and the resulting setpoint changes incorporated in the TS for the extended power uprate are consistent with the Hatch licensing basis and are, therefore, acceptable.

3.0 DESIGN BASIS ACCIDENT RADIOLOGICAL CONSEQUENCES

The licensee reevaluated the radiological consequences of the following three postulated DBAs at an uprated reactor power level of 2818 MWt (102 percent of extended uprated reactor power of 2763 MWt) in Section 9.3 of the license amendment request (Reference 1) and in a separate submittal dated April 17, 1997 (Reference 8). The analyzed DBAs are (1) LOCA, (2) Fuel Handling Accident, and (3) CRD Accident. The licensee provided additional information on radiological consequence analyses (Reference 3) in response to the staff's request. The licensee stated in the amendment request, and the staff agrees, that the main steamline break accident outside containment was not reanalyzed because the mass flow rate from the postulated main steamline break is unchanged from the original power uprate. The licensee concludes that the radiological consequences of an accident subsequent to implementation of the Hatch power uprate remain below the dose criteria specified in 10 CFR Part 100 and General Design Criterion (GDC) 19 of Appendix A to 10 CFR Part 50.

The staff reviewed the radiological consequence analyses performed by the licensee and finds that the calculational methods used for the radiological consequence assessments are acceptable, with the following comments on the use of the computer code ARCON95 to determine the control room relative concentration (X/Q) values. The licensee has calculated the control room X/Q values using calendar year 1995 hourly meteorological data and the ARCON95 methodology described in NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes." The licensee has calculated X/Q values for various postulated ground level release point and intake pairs. Elevated release calculations were also made for a postulated release from the 120 meter plant stack.

The ARCON95 methodology calculates plume centerline X/Q values for the 0 to 2-hour time period following the beginning of the postulated accident and sector average X/Q values for the remaining time of the accident. The centerline concentration should be calculated for the 0 to 8-hour time period following initiation of the accident. Therefore, the staff made calculations of the centerline X/Q for the 0 to 8-hour time period using the ARCON96 methodology and compared the results with the licensee's calculations. The staff has concluded that, for this specific accident scenario, the impact on the total resultant dose in using the highest X/Q centerline concentration only for the 0 to 2-hour time period is not significant.

The licensee has used calendar year 1995 hourly meteorological data in its assessment of the control room X/Q values. Use of a single year of data could result in uncertainties due to year-to-year variabilities in meteorological conditions. However, the licensee performed a comparison and provided summaries of historic plant data to demonstrate that the data used in its calculations of the X/Q values are adequately representative of long-term conditions. Winds at the 10 meter level at the Hatch site appear to be lighter than expected when compared with expectations based on winds at the 60 meter and 100 meter measurement levels. However, the staff has concluded that for this specific accident scenario, the impact on the total resultant dose is not significant.

Therefore, the staff finds the use of the control room X/Q values (listed in Table 4) proposed by the licensee to be acceptable. The licensee should, however, use a methodology that calculates the centerline relative concentrations for the 0 to 8-hour time period when making any additional calculations of control room X/Q values in the future.

To verify the licensee's conclusion, the staff performed an independent radiological consequence analysis using that performed by the staff for the LOCA in License Amendment No. 132 (Reference 16), which was analyzed at a reactor power level of 2537 MWt. The staff believes that the radiological consequences of an accident subsequent to implementation of the Hatch extended power uprate will be increased approximately proportional to the increase in reactor power. For the potential radiological consequence to the main control room operator, the staff used new control room X/Q values proposed by the licensee.

Based on its evaluation and the licensee's analyses, the staff concludes that the radiological consequences for the requested extended power uprate still remain within the relevant dose criteria and there is reasonable assurance that the radiological consequences of bounding DBAs will not exceed dose acceptance criteria specified in the SRP, 10 CFR Part 100, and GDC-19 of Appendix A to 10 CFR Part 50. The dose acceptance criteria are 25 rem whole body or 300 rem to the thyroid at the EAB or LPZ and 5 rem whole body or 30 rem thyroid to the control room operator. Therefore, the staff finds the proposed extended power uprate to be acceptable. The resulting radiological consequence analyses are provided in Table 3. The control room and site boundary X/Q values used by the staff are provided in Tables 4 and 5, respectively.

4.0 HUMAN PERFORMANCE

The staff's evaluation, which was composed of five review topics, is provided below.

Topic 1 - Discuss whether the extended power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the extended power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will it require any new operator actions?

By letter dated August 8, 1997, the licensee stated that the extended power uprate would not change the type or scope of plant emergency and abnormal operating procedures. The licensee also stated that the extended power uprate would not change the type, scope, or 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 the extended power uprate and address whether the extended power uprate will have any effect on operator reliability or performance. Identify operator actions that would necessitate reduced response times associated with the extended power uprate. Please specify the expected response times before the extended power uprate and the reduced response times. What have simulator observations shown relative to operator response times for operator actions that are potentially sensitive to the extended power uprate? Please state why reduced operator response times are needed. Please state whether reduced time available to the operator due to the extended power uprate will significantly affect the operator's ability to complete manual actions in the times required.

By letter dated March 9, 1998, the licensee stated that operator action will remain unchanged as a result of implementing the extended power uprate and that simulator observations have shown no noticeable reduction in effectiveness of operator response. By letter dated September 11, 1998, the licensee stated that it evaluated the impact of the extended power uprate on the time available to the operator to depressurize the reactor under a non-ATWS condition. The results of the licensee's evaluation are summarized in the table below.

Power Level	Operator Time Available tc Initiate Depressurization (in minutes)
Orignial power level (2436 MW_t)	1.1
Extended power uprate (2763 MW_t)	1.2

The table shows that the time for the operator to initiate depressurization at the extended power uprate power level was 0.1 minute longer than the time at the original power level. On the basis of the insignificant difference in the change in operator response time to initiate depressuization, the staff finds that the extended power uprate does not affect the operator's ability to initiate depressurization. The staff also finds that the licensee has provided satisfactory responses to Topic 2.

Topic 3 - Discuss any changes the extended 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 letter dated March 9, 1998, the licensee stated that the extended power uprate will have minimal impact on control room instruments and controls. The licensee noted that no changes to front panel indicators or controls are required. The licensee indicated that the following changes would be implemented: new programmable controls for the recirculation pump runback and the ranges for the main steamline flow transmitters, which send signals to the control panel indicators. The staff finds that the licensee's responses are satisfactory since the subject changes are enhancements to the control room instruments and controls.

Topic 4 - Discuss any changes the extended power uprate will have on the Safety Parameter Display System (SPDS).

By letter dated March 9, 1998, the licensee stated that digital signal input and output lists for the SPDS would not be affected. The licensee also stated that extended power uprate changes (i.e., setpoints and calibration) to field devices that provide a signal to the computer will be automatically reflected in the SPDS. The staff finds that the licensee's responses are satisfactory.

Topic 5 - Describe any changes the extended power uprate will have on the operator training program and the plant simulator. Provide a copy of the post-modification test report (or test abstracts) to document and support the effectiveness of simulator changes as required by ANSI/ANS 3.5-1985, Section 5.4.1.

(a) Provide classroom and simulator training on the extended power uprate modification.

The licensee stated in its letter of March 9, 1998, that an operator training lesson plan will be prepared to teach plant changes as a result of the extended power uprate and that existing lesson plans will be flagged for revision during the next regular revision cycle. The extended power uprate lesson plan will be presented to all eligible licensed/certified shift personnel before plant startup for extended power uprate operation and to all licensed/certified personnel during the following segment of requalification training. Additional training regarding the extended power uprate will be incorporated in continuing training lesson plans for other training sections, as applicable.

(b) Complete simulator changes that are consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be revalidated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." Simulator revalidation will include comparison of individual simulated systems and components and integrated plant steady-state and transient performance with reference plant responses using similar startup test procedures.

The licensee stated in its letter of March 9, 1998, that simulator changes will be implemented before plant startup for extended power uprate operation. Simulator revalidation will be accomplished in two stages. First, the simulator performance will be validated against the extended power uprate expected system response. Second, post-startup data will be collected and compared with simulator performance data, allowing any necessary adjustments to simulator model performance. This simulator performance validation will be performed in accordance with ANSI/ANS 3.5-1985, Section 5.4.1.

(c) Complete control room and plant process computer system changes as a result of the extended power uprate.

See evaluation of Topics 3 and 4.

(d) Modify training and plant simulator relative to issues and discrepancies identified during the startup testing program.

The licensee stated that the simulator performance discrepancies will be identified during the extended power uprate power ascension testing through comparison of plant data with simulator performance data.

On the basis of the information discussed, the staff finds that the licensee has proposed satisfactory changes to the operator training program and the plant simulator as a result of the extended power uprate.

In summary, the staff concludes that the previously discussed review Topics 1 through 5 associated with the proposed extended power uprate for Hatch have been or will be satisfactorily addressed. The staff further concludes that the extended power uprate should not adversely affect operator performance or operator reliability.

5.0 ELECTRICAL AUXILIARY DISTRIBUTION SYSTEM

The staff has reviewed the information submitted by the licensee of the BOP licensing report to determine if the extended power uprate would have an adverse impact on the station's electrical auxiliary distribution system and to see if a TS revision would be necessary.

5.1 Summary of Plant Modifications

The licensee has reviewed the potential impact of the extended power uprate on the main generator stator and isophase bus cooling, temperature-monitoring system in main transformers, adjustment to the main generator, and switchyard main protective devices. In response to the staff's RAI, the licensee submitted the following details regarding the current plan for modifications.

Since the extended power uprate will increase generator rating from 1000 megavars (MVA) to 1050 MVA at 0.85 and 0.88 power factor, respectively, the generator will require an upgrade of the stator water cooling system. The plant service water flow rate to the stator water cooling system will be increased from 1750 gallons per minute (gpm) to 1880 gpm at 95 °F to remove the additional heat from the stators due to increase in the generator rating, and the stator bar flow rate will increase to 550 gpm. The stator bar flow rate will be increased by replacing the main flow orifice and main filter. Also, the flow meter will be replaced with one having a calibration span compatible with the new flow element. Setpoints for the alarm and interlocks in the stator water cooling system will be changed to accommodate the new system's operating parameters (flow, pressure, and temperature).

The licensee also assessed the impact of an increase in steam flow corresponding to a 113 percent extended flow uprate from the original design on the nonsafety-related, isophase bus duct and its cooling equipment. The licensee determined by field testing that no major changes need be made to the Unit 1 isophase bus duct cooling system to support the extended power uprate, because the changes in Unit 1 are very minor.

However, the Unit 2 generator isophase bus duct cooling system will require an upgrade to accommodate the extended power uprate. The plant service water flow rate to the isophase bus duct cooling system will be increased from 156 gpm to 160 gpm at 95 °F to remove the additional heat from the isophase bus duct. The cooling coils, the fans, and the fan motors will be upgraded. The duct work will be modified to accommodate the larger fans and cooling coils. Instrumentation in the duct will be replaced. To accommodate the increased plant service water flow, the plant service inlet and outlet piping diameter to the cooling coils will be increased from 2 ½-inch to 3-inch diameter, and an orifice in the discharge piping will be removed or resized. The inlet and outlet carbon steel piping will be upgraded to a more corrosion-resistant material. Also, the aluminum bars used as connectors at the generator will be replaced with flexible copper braided connectors to improve reliability and reduce maintenance.

The transformers are adequately rated to support power uprate operation. The Unit 1 main transformer rating is 1008 MVA. The maximum load on the transformer at maximum power levels is 954 MVA, which is well within the rating of the transformer. Therefore, no supplemental temperature monitoring is needed. However, the Unit 2 transformer rating is 997.8 MVA. The maximum load on the transformers at the maximum power level is 991 MVA, which is close to the rating of the transformer. A new temperature-monitoring system will be installed in the Unit 2 main transformer to provide accurate data for winding and oil temperature during the power ascension program. Plant operation personnel can also use the temperature-monitoring system to provide more accurate data for determining transformer performance during various degrees of electrical loading with respect to operating conditions such as ambient temperature or loss of cooling fans.

During the extended power uprate program, the main generator controls and switchyard devices will require adjustment or replacement of some components for proper operation of the equipment. The licensee concluded that the controls and devices stated below will be impacted, and the following adjustments are planned:

(1) The generators are capable of meeting the proposed uprate without undergoing physical modifications to the generator, coolers, or excitation system hardware. However, the licensee proposed resetting the underexcited reactive ampere limit (URAL) and annunciator/control for excitation system of the generators. A new set of generator performance curves, including excitation V-curves, reactive capability curves, and saturation curves, were provided as plant-specific instruction book updates. The information contained in these curves will be programmed into the turbine generator and excitation control systems during the refueling outage. These programming changes will allow operation at the new level of reactive ampere limits, annunciator setpoints, and various other settings resulting from the extended power uprate generator rerating.

(2) The setpoints and controls associated with the stator water cooling runback logic will be changed as a result of the extended power uprate.

(3) The 3000/5 amp current transformers (CTs) for the Offerman Line are currently tapped at 1200/5 amp. During extended power uprate operations, these CTs could see current in excess of 1200 amp if the 230 kV switchyard breaker 510 (between the Unit 1 generator and 230 kV Bus 1) is out for maintenance or is tripped. Therefore, the CT ratio taps will be changed from 1200/5 amp to 2000/5 amp.

(4) As a result of CT ratio tap changes for the extended power uprate, 12 protection relays for the 230 kV Offerman Line and two ground detectors associated with 230 kV switchyard breaker 510 and breaker 490 will be reset.

(5) As a result of CT ratio tap changes, the existing IRQ-9 directional ground relay will be replaced with a new relay. The existing relay has a setting of 12 amp with an instantaneous range of 10 amp to 40 amp. The replacement relay will have the setting of 7 amp with an instantaneous range of 4 amp to 16 amp.

On the basis of the information presented by the licensee, the staff reviewed a summary of plant modifications to be implemented by the licensee preceding the proposed power uprate. None of these modifications are safety-related and no TS changes are required; therefore, they are acceptable to the staff.

5.2 GDC-17 and Station Blackout

The licensee reviewed and evaluated whether the extended power uprate would alter the original licensing basis for GDC-17 and the SBO requirements.

5.2.1 Electrical Power and Auxiliary System

The licensee evaluated onsite and offsite electrical supply and distribution systems for safety-related components in conformance to GDC-17 (10 CFR Part 50, Appendix A). The significant results of the evaluation follow.

5.2.2 Generation and Offsite Power System

(1) The cooling system for the main electrical generator stator has been enhanced to increase the MVA rating of the main generator for the extended uprate power level.

(2) With cooling system improvements to the Unit 2 isophase phase bus duct, it will have adequate capacity to carry the maximum generator full load current at the nominal voltage of 24 kV and at the minimum voltage of 22.8 kV under postulated worst-case loading conditions.

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

(4) The plant stability scenarios outlined in the FSARs were evaluated at the extended uprate power level. The licensee concluded that there are no adverse effects on grid stability or plant reliability.

(5) System grid load flow studies were performed using projected 1998 peak and valley load conditions with the Hatch units at the extended uprate power level for scenarios outlined in the FSARs. The licensee concluded that resultant switchyard voltages were adequate to achieve safe shutdown from the offsite power network.

In response to the staff's RAI, the licensee presented details about how the analysis was performed and the assumptions that were used. In response to a question about system grid load flow studies, the licensee elaborated that under the worst-case condition (i.e., the loss of the 500-kV line with a subsequent LOCA in either unit, assuming a maximum 1-hour demand peak load during the summer season and the grid postulated to deliver the maximum guaranteed demand of 3600 MWe of power to Florida, which results in the lowest voltage in the Hatch switchyard), either unit will remain within the design capability curve of the main generator and can sustain the integrity of the offsite power system network for LOCA mitigation. During postulated grid upset conditions, system dispatchers will take compensatory measures to restore the grid voltages by turning on capacitor banks and other appropriate equipment within minutes of the event to maintain long-term required grid voltage. This will be in accordance with their prescribed voltage schedule, which includes guaranteed minimum switchyard voltages for the Hatch units. The licensee concluded that the resultant switchyard voltages after the extended power uprate will be adequate to safely shut down the units from the offsite network, and the staff finds this acceptable.

5.2.3 Onsite Power Distribution System

Station loads under normal operation/distribution conditions are computed on the basis of equipment nameplate data. Since the extended power uprate does not require equipment operation above nameplate rating, the electrical supply and distribution components (switchgear, MCCs, cables, etc.) are adequate.

Station loads under emergency operation/distribution conditions (diesel generators) are based on equipment nameplate data. The extended power uprate will not change the power requirements of any safety-related load; therefore, under emergency conditions, the electrical supply and distribution components are adequate.

The dc power distribution system provides control and motive power for various systems and components within the plant. System loads are computed on the basis of equipment nameplate data. Operation at the extended uprate power level will not increase any loads beyond the nameplate rating or revise any control logic; therefore, the dc power distribution system is adequate.

Even though no equipment replacement was necessary for the extended power uprate that would increase electrical loads beyond the design ratings or above levels previously analyzed, the licensee has reassessed the adequacy of the offsite and onsite power distribution system analysis to ensure that with the increase in Hatch generation output, it would remain in conformance with GDC-17. The licensee finds that the safety functions of the offsite and onsite electric power systems are not affected by the uprated conditions; therefore, extended power uprate has no effect on the plant's conformance with GDC-17 requirements.

On the basis of the licensee's preceding evaluation, the staff finds that the plant shutdown equipment will continue to perform its intended safety-related functions for the extended power uprate. The staff concludes that the station auxiliary electrical distribution system is not adversely impacted by the proposed extended power uprate at Hatch.

5.2.4 Station Blackout

The licensee reviewed the SBO analysis and observed that plant response and coping capabilities for the SBO event are affected slightly by operation at the extended power level because of the increase in the decay heat. There are no changes to the systems and equipment used to respond to an SBO event, nor is the required coping time changed. The Hatch coping duration for SBO is 4 hours. However, suppression pool cooling, which uses the alternate AC (AAC) power source, can be initiated in 1 hour when the diesel leading margins are met.

The licensee also observed that "As part of the extended power uprate, the SBO scenario was reanalyzed assuming that the suppression pool cooling (SPC) was initiated in one hour when the AAC power source is assumed available. The peak pool temperature is 167°F. Even if SPC is not initiated until 4 hours, the resulting peak pool temperature of 206°F is acceptable for containment and ECCS pump operation." In response to this observation, the staff, in an RAI, asked the licensee to clarify whether credit is being taken for suppression pool cooling availability within 1 hour or within 4 hours. The preceding information appears to indicate that the AAC power source (Emergency Diesel Generator (EDG) 1B), will be available to provide SPC 1 hour after SBO and to limit the pool temperature to 167 °F. However, in a May 3, 1991, response to a staff concern that EDG 1B would be overloaded unless operators shed loads during an SBO, the licensee told the staff that the completed analyses demonstrated that SPC was not required during the 4-hour SBO coping duration. In response to the RAI, the licensee also added the following: "The fact that the SBO analysis shows SPC is not required for 4 hours does not preclude using the EDG to power SPC within 1 hour if the diesel loading margins are met and the operator so chooses. There is no change to the safety-related loads that would be powered by EDG 1B due to extended power uprate. Therefore, the existing assumptions related to 1 hour initiation of SPC and the acceptability of the 4-hour coping period are no different than those documented in the original SBO submittal, which was approved by the NRC in the SER dated November 1, 1991."

The staff reviewed the original SBO submittal and SE dated November 1, 1991, and concluded that the assumptions used in the extended power uprate analysis did not differ from assumptions made in the SBO analysis that the staff reviewed and accepted. The staff concludes that the SBO analysis is acceptable for the extended power uprate.

6.0 EQUIPMENT QUALIFICATION

The licensee reviewed the impact on safety-related electrical equipment qualification for the extended power uprate for normal and accident conditions inside and outside of containment. Applicable conservatism in accordance with IEEE Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," was applied to the environmental parameters as required.

The licensee observed that, during an accident at extended power uprate conditions, the maximum humidity will remain at 100 percent. The maximum pressure will increase to less than 51 psig but will not affect qualification. The accident temperature profile at the extended power uprate condition exceeds the current accident profile by up to 7 °F during the time period from 35,000 seconds to 70,000 seconds. This will have no effect on qualification of any equipment. The extended power uprate accident radiation dose has been adjusted for the original power uprate of 5 percent already implemented and the proposed extended power uprate of 8 percent. In response to the staff's RAI concerning whether, for each component on the equipment qualification (EQ) Master List, the existing qualification test data envelope the accident temperature profile at extended power uprate conditions with the required margin, the licensee furnished curves showing Unit 1 and 2 drywell temperature EQ profiles. The profiles showed that the peak drywell temperature under worst-case accident conditions is below 330 °F, which is the peak temperature presently assumed in evaluating the adequacy of environmental qualification tests for drywell equipment. The licensee also stated in the response that degraded equivalency analysis, documented in calculation SINH 97-004, shows that the present worst-case, design-basis event profile envelopes the new accident profile at extended power uprate conditions. The staff reviewed the calculation and concludes that the present profile has enough conservatism to envelope the new accident profile at the extended power uprate conditions. However, the degraded equivalency analysis is based on the Arrhenius Methodology and, as a separate and generic issue outside the scope of power uprate reviews, the staff will evaluate the acceptability of the Arrhenius Methodology for environmental qualification under LOCA and post-LOCA conditions.

The licensee also stated that a review of equipment qualification at the extended power uprate conditions identified some equipment located inside containment that is potentially affected by the higher than normal radiation level. The qualification of this equipment has been reevaluated on the basis of location-specific dose calculations. This equipment has been found acceptable for the extended power uprate conditions, although it will be necessary to decrease the qualified life of specific components because of increased radiation levels during the extended power uprate conditions. In an RAI, the staff asked the licensee to provide a detailed discussion of the equipment potentially affected and to quantify the decrease in qualified life for potentially affected equipment.

In response to the RAI, the licensee furnished the following details:

The environmental qualification review for the extended power uprate consists of: (1) an initial screening of all components on the EQ Master List for the effect on qualification parameters such as temperature, pressure, and radiation; (2) a location-specific dose calculation for the equipment identified in the initial screening as potentially impacted; and (3) a reevaluation of all equipment on the EQ Master List based on the results of the location-specific dose calculation.

Radiation was determined to have the greatest potential impact on equipment qualification. The initial screening assumed an 8 percent increase in the specified 40-year total integrated dose (TID) inside the drywell. From the initial screening, Target Rock solenoid valves (model 1/2 SMS-S-02-4) and NAMCO limit switches (model EA180 and EA740) located inside the drywell were identified as being potentially impacted. A location-specific calculation was then prepared for these devices.

The calculation demonstrated that the Target Rock solenoid valves inside the drywell are qualified for 40-year TID at the extended power uprate conditions. The calculation also identified five NAMCO limit switches with 40-year, location-specific doses that will exceed the tested dose of 5.0E+07 rads. Therefore, the calculation included a determination of revised qualified life based on radiation.

Temperature is currently the most limiting factor in determining the qualified life of the NAMCO limit switches. The thermally-qualified life remains the most limiting for three of the limit switches; however, the radiation-qualified life will be most limiting for two of the limit switches for the extended power uprate.

The calculation also determined a revised bounding maximum dose for any equipment in the drywell. The qualified life of all EQ components in the drywell was reviewed with respect to a revised maximum bounding dose. No additional equipment was affected. The replacement intervals for affected equipment will be revised upon implementation of the extended power uprate.

On the basis of the information submitted by the licensee, the staff concludes that the qualification of safety-related electrical equipment has been adequately addressed and the equipment will continue to be qualified for the extended power uprate.

6.1 EQ of Mechanical Equipment with Nonmetallic Components

The licensee evaluated the effect of the extended power uprate on the nonmetallic parts of equipment and components, such as pumps and heat exchangers. The structures, systems, and components (SSCs) that could be impacted by changes associated with the extended power uprate were identified. The licensee evaluated the changes in system pressures, temperatures, and flow rates, and determined that most of the SSCs were within their original design capabilities with no additional actions needed. The licensee stated that if a design change to any SSC was necessary to assure the compatibility with normal or accident conditions, then the changes would be performed under the Quality Assurance Program at SNC. Therefore, the licensee concluded that the FSAR design and qualification requirements of nonmetallic parts of equipment and components would continue to be met at extended power uprate conditions.

Based on its review, the staff finds that the environmental qualification of the nonmetallic components at the extended power uprate conditions is acceptable under the administrative controls of SNC's Quality Assurance Program pursuant to Appendix B to 10 CFR Part 50 and 50.34(b)(6)(ii).

7.0 INDIVIDUAL PLANT EXAMINATION (IPE)

The licensee addressed the impact of the proposed Hatch extended power uprate on plant risk. The proposed extended power level of 2763 MWt represents an 8 percent increase in reactor thermal power from the current power level of 2558 MWt. The licensee had received its first power uprate approval of 5 percent increase (2436 MWt to 2558 MWt) from the staff on August 31, 1995. The Hatch IPE was performed prior to this first power uprate approval. The licensee stated that the original power uprate was judged to have had a negligible impact on the calculated core damage frequency (CDF). The probabilistic risk assessment (PRA) performed for the proposed extended power uprate accounts for the combination of both the first power increase and the proposed extended increase. Thus, this analysis considered a 13 percent difference in the power level with the baseline PRA representing the plant at 2436MWt and the extended power uprate PRA at 2763MWt.

7.1 Evaluation

The staff reviewed Section 10.5 "Individual Plant Evaluation" of the licensee's submittal. The staff's evaluation addressed the licensee's discussion on risk pertaining to internal events (Level 1), containment analysis (Level 2), and external events (fire and seismic).

7.1.2 Internal Events

The licensee addressed PRA attributes that could potentially be affected by the proposed extended power increase. These included: initiating event frequencies, success criteria, and operator actions. In addition, the licensee also addressed the effect of the extended power uprate on component failure rates in Reference 4.

The licensee's analysis considered the impact of 13 percent power increase on various accident initiating events. These internal initiating events included: (1) LOCA -- small, medium, large, and LOCAs outside the containment, (2) anticipated transients -- reactor scram, turbine trip, loss of feedwater, loss of condenser vacuum, MSIV closure, inadvertent opening of a relief valve, loss of drywell cooling, loss of control room cooling, loss of plant service water, loss of various ac and dc busses, loss of grid during various initiators, (3) loss of offsite power (including station blackout), and (4) ATWS -- MSIV closure, loss of feedwater, and turbine trips. The licensee performed a qualitative review of the underlying contributors to these initiating events to determine the potential effects of the extended power uprate on the initiating event frequencies. The licensee concluded that the extended power uprate would have no readily discernible adverse effect on initiator frequency. The staff finds it reasonable to conclude that initiating event frequencies would not be changed as long as operating band/limits of equipment are not exceeded. However, the staff notes that if any change in the initiating event frequencies is observed in the future, it would be tracked under the plant's Maintenance Rule program.

The licensee determined that the critical functions in maintaining plant conditions to prevent core damage are: reactivity control, pressure control, high pressure inventory control, vessel depressurization, low pressure inventory control, and long-term containment heat removal. To evaluate the success criteria for these functions, the licensee performed a series of Modular Accident Analysis Program (MAAP) calculations which re-computed the plant's thermal-hydraulic performance and timing of events at the proposed higher power levels. The results of these calculations were used to obtain certain plant parameters during transients or plant accidents. The information was then compared to the specific success criteria of the baseline PRA. The licensee determined that the success criteria for the extended power uprate PRA did not change from the success criteria for the baseline PRA and that they remain adequate for the extended power uprate level. The staff finds this conclusion acceptable.

With respect to the potential impact on equipment reliability by the proposed extended power uprate, the licensee reported that the component failure rate will not change significantly with the extended power uprate because the component monitoring programs that are in place, e.g., environmental qualification, and erosion/corrosion, will be modified to account for the extended power uprate. The licensee stated that the extended power uprate may result in components being refurbished or replaced at more frequent intervals; however, the functionality and reliability of components will be maintained to the current standard. Similar to the above discussion on the effect of the extended power uprate on initiating event frequency, the staff finds it reasonable to conclude that component failure rate would not be changed as long as operating band/limits of component are not exceeded. However, the staff notes that if any change in the equipment failure probability is observed in the future, it would be tracked under the plant's Maintenance Rule program.

The licensee conducted an evaluation to determine how the proposed extended power uprate would impact operator response capabilities during accidents. When shorter time is available for an operator to diagnose and execute an action in response to an accident situation, a higher human error probability would be estimated for that action. The licensee identified operator actions for which the reduction in time available to complete the action could result in significant change in risk. These operator actions included: failure to depressurize given inadequate high pressure injection (non-ATWS), failure to depressurize given inadequate high pressure injection (ATWS), failure to initiate Standby Liquid Control with Turbine Bypass Valve capacity unavailable (ATWS), failure to control low pressure injection after depressurization (ATWS), and grid recovery probability for station blackout without high pressure injection.

Sensitivity studies of these operator actions were performed to evaluate their impact on risk. The results of the sensitivity studies showed that operator actions involved in ATWS sequences did not have a significant impact on CDF. The result of calculating grid recovery probability for station blackout on the basis of new shorter available time showed that the recovery probability did not change significantly and the resulting impact on the CDF was determined to be negligible. However, the operator action to depressurize the reactor (non-ATWS) was shown to have a potentially significant impact on the CDF.

This operator action is not the only operator depressurization action in the PRA models. It is specific only to those non-ATWS events requiring depressurization during situations in which the operators have to resort to estimating the vessel water level due to potential incorrect level indications caused by elevated drywell temperatures or overheated main control room instrumentation. This is the reason that most of the sequences impacted by this operator action involve medium-LOCA or loss of main control room cooling initiating events.

To evaluate the impact on the time available to accomplish this operator action and the related effect on the operator failure probability, the MAAP analyses were conducted. The MAAP analyses demonstrated that the change in operator response time for this particular action was negligible; however, the licensee performed a bounding analysis by postulating the operator failure probability to have increased by a factor of two. The resulting CDF from internally initiated events was about 2.3×10^{-5}/year for the proposed extended power level for Hatch Unit 1 and about 2.4×10^{-5}/year for Unit 2. This represents an increase of about 1×10^{-6}/year from the baseline CDF of 2.2×10^{-5}/year for Unit 1 and an increase of about 1×10^{-6}/year from the baseline CDF of about 2.3×10^{-5}/year for Unit 2.

Based on the reported analysis and results, the staff finds that the change in CDF from internal events due to the proposed extended power uprate is small and is acceptable, and that this increase is mainly due to a postulated increase in one human error probability.

7.1.3 Level 2 Internal Events PRA

The licensee's evaluation showed that there were no new PRA sequences in the Level 1 reanalysis, that required specific evaluation to determine the impact on containment performance. That is, the extended power uprate did not modify the PRA model sequence database to an extent that would alter the baseline PRA results. The licensee found that, in general, the operator actions and success paths for Level 2 associated with this analysis were sufficiently long-term so that they would not be significantly affected by the proposed extended power uprate.

In response to the staff's RAI dated March 27, 1998, which requested that the licensee provide any quantitative results for Level 2 analysis, the licensee reported that an approximate 1 percent change in CDF was noted for those sequences that were part of the Large Early Release Frequency (LERF) for each unit when evaluated for the extended power uprate. This was considered a very small change. The revised probability for operator failing to depressurize the reactor (non-ATWS) introduced no new sequences to be addressed in the Level 2 analysis. Thus, the licensee maintained that the containment analysis performed for the baseline PRA was still valid for the extended power uprate case. The staff finds that the impact of the proposed extended power uprate on LERF is very small.

7.1.4 Internal Fire, Seismic, and Other External Events PRA

The licensee estimated that the CDF associated with fire increased by less than 1 percent from the baseline of 5.6×10^{-6}/year for Hatch Unit 1 and approximately 2.2 percent from the baseline of 3.4×10^{-6}/year for Hatch Unit 2. These increases, less than 6×10^{-8}/year for Unit 1 and about 8×10^{-8}/year for Unit 2, were attributed primarily to the operator action of failing to depressurize given inadequate high pressure injection (non-ATWS).

With respect to seismic events, the licensee reported that the proposed extended power uprate has no effect on the seismic margins analysis (SMA) performed for Hatch. The increase in the power level is not expected to affect equipment survivability nor equipment response during an earthquake, nor does it modify the safe shutdown pathway assumed in the SMA. The staff considers the risk increase from internal fire and seismic events due to the proposed extended power uprate to be small and is acceptable.

7.1.5 Shutdown Risk

The licensee did not perform a quantitative analysis of the proposed extended power uprate's impact on shutdown risk since a Hatch shutdown risk model is not available. Instead, the licensee examined the shutdown risk impact in a qualitative manner by addressing shutdown risk-related questions posed in the staff's Standard Review Plan, Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance."

The questions in the SRP pertained to the proposed extended power uprate's impact on shutdown schedule, operator ability to take actions, shutdown equipment reliability, and availability of equipment or instrumentation used for contingency plans. For a potential impact on shutdown risk, the licensee determined that although it would take longer to reduce temperature and pressure to closed cycle RHR entry conditions, and to achieve cold shutdown, there would be very little change to the shutdown schedule, and no direct safety impacts on the schedule. The licensee determined that although an increase in decay heat would result in a small decrease in the time available for operator actions during shutdown, operators have many hours, or days, during cold shutdown so that any small decrease in time for operator action would not be a critical factor in the human error probability. The licensee also concluded that equipment reliability will be maintained to the current standards because the equipment is expected to be operated within acceptable design and operational limits. Lastly, the licensee determined that no impact on equipment availability or instrumentation used for contingency plans is expected due to the proposed extended power uprate. The staff finds that the preceding factors are not impacted significantly by the proposed extended power uprate and considers any related risk increase to be negligible.

7.1.6 Uncertainty

To address parameter uncertainties, the licensee performed an analysis to propagate the distribution representing uncertainty on the basic parameter values to generate a probability distribution on the CDF and LERF of the Hatch PRA. The results indicated that the shapes of the uncertainty distributions are virtually unchanged between the original power level case and the proposed extended power uprate case. The estimated values of the median and the 95th and 5th percentile upper and lower bounds showed a small shift in the uncertainty distribution to reflect the small increase in the mean CDF and LERF, but the error factors remained essentially unchanged at about 2.3 for CDF and about 4 for LERF. The staff finds that these error factors indicate reasonable ranges and that parametric uncertainty is fairly low.

As previously discussed, the extended power uprate changes primarily affect timing of operator actions, and do not affect the types of initiators, sequences, success criteria, systems models, or plant configuration and operation. Therefore, the licensee's conclusion that model uncertainty is not significantly impacted by the proposed extended power uprate is acceptable.

7.1.7 Quality of PRA

The staff believes that the review of the quality of the PRA should be commensurate with the role that the PRA results play in the staff's decision process and with the degree of rigor needed to provide a valid technical basis for the staff's decision. In this case, the licensee is not requesting relaxation of any deterministic requirements for the proposed extended power uprate application, and staff approval is based on the licensee meeting the current deterministic requirements. Therefore, the staff considered its original evaluation of the Hatch Individual Plant Examination,[1] the PRA portion of the extended power uprate submittal, and additional information provided to the staff constitute sufficient indication of the PRA quality.

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[1] The licensee's original IPE was submitted to the NRC in 1992 and the Staff Evaluation Report (SER) accepting the submittal was issued by the staff in 1995. As stated in the SER, the staff found that (1) the Hatch IPE was complete with respect to the information requested in Generic Letter 88-20 and associated supplement 1, (2) the analytic approach was technically sound and capable of identifying plant-specific vulnerabilities, including those associated with internal flooding, (3) the licensee employed a viable means to verify that the IPE models reflect the current plant design and operation at time of submittal to the NRC, (4) the IPE had been peer reviewed, (5) the licensee participated in the IPE process, (6) the IPE specifically evaluated the decay heat removal function for vulnerabilities, and (7) the licensee responded appropriately to the Containment Performance Improvement program recommendations. Based on these findings, the staff concluded that the licensee met the intent of Generic Letter 88-20.

7.2 Conclusion

Based on the reported analysis and results, the staff finds that the increase in CDF (from internal, external, and shutdown events events) and LERF due to the proposed extended power uprate is small and that this change is mainly from a postulated increase in one human error probability, namely operator failing to manually depressurize the reactor. The staff finds that, based on the current analysis, no significant change can be predicted for initiating event frequencies, success criteria and component failure rates. The staff also reviewed the results of the licensee's uncertainty analysis and agrees that the proposed extended power uprate does not have a significant impact on the parametric and modeling uncertainties. In conclusion, the staff finds that the impact of the proposed extended power uprate on plant risk is small.

8.0 REGULATORY COMMITMENTS

In the submittals dated March 9, 1998, and July 31, 1998, the licensee committed to do the following:

1. Perform cycle-specific fuel thermal-mechanical limit evaluations (Section 2.1.b)

2. Evaluate and verify the acceptability of the results of the plant-specific LOCA analysis at each reload (Section 2.4)

3. Notify the NRC if the required containment overpressure increases by 1 foot (Section 2.8.3)

4. Provide classroom and simulator training (addressed - see 10 CFR 50.120 and 10 CFR 55.59) and complete simulator and computer system changes (Section 4.0)

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of the proposed changes pertaining to the preceding regulatory commitments are best provided by the licensee's administrative processes, including its commitment management program. The preceding regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes). The staff notes that pending industry and regulatory guidance pertaining to 10 CFR 50.71(e) may call for some information related to the preceding commitments to be included in a future update of the facility's FSAR.

9.0 SUMMARY

Based on the preceding safety evaluation, the staff concludes the operation of Hatch, Units 1 and 2, at the extended power uprate level of 2763 MWt is acceptable. Accordingly, the proposed change to paragraph 2.C.(1) of both licenses and the related TS changes are acceptable, and will not adversely impact the safe operation of the plant at the increased power level and there is a reasonable assurance that the radiological consequences of normal operations and anticipated accidents at the increased power level will continue to meet the applicable regulatory requirements.

10.0 EXIGENT CIRCUMSTANCES

The Commission's regulations, 10 CFR 50.91, contain provisions for issuance of amendments when the usual 30-day public notice period cannot be met. One type of special exception is an exigency. An exigency is a case where the Commission and licensee need to act promptly and time does not permit the Commission to publish a Federal Register notice allowing 30 days for prior public comment, and it is determined that the amendments involve no significant hazards consideration.

Under such circumstances, the Commission notifies the public in one of two ways: by issuing a Federal Register notice providing an opportunity for hearing and allowing at least 2 weeks for prior public comments, or by using the local media to provide reasonable notice to the public discussing the proposed changes. In this case, the Commission used the first approach.

The licensee submitted the proposed changes by letter dated August 8, 1997. In processing this request, the staff recognized on September 29, 1998, that it inadvertently failed to publish a notice of proposed issuance of the amendments in the Federal Register. In the August 8, 1997, original application, the licensee requested that the proposed amendments be issued prior to startup from the fall 1998 refueling outage on Unit 2. Startup from the refueling outage is presently scheduled for October 18, 1998.

Upon being informed by the staff that a notice of proposed issuance of amendments inadvertently was not published, the licensee requested, by letter dated September 30, 1998, that the proposed amendments be processed on a exigent basis.

The need for exigency is based on the fact that the licensee would be required to postpone changes to procedures, instrumentation, and setpoints on Unit 2 until after startup and power ascension of the plant if the amendments were not issued prior to restart. The licensee would then be required to implement these changes while online which would increase the possibility of a plant scram and introduce a potential for unnecessary transients on the plant.

The licensee has evaluated the impact of the schedule change and the online implementation of the extended power uprate (EPU) and determined that receiving the amendments prior to startup will result in a net increase in plant safety and reliability. Reliability benefits include a reduced potential for an inadvertent reactor scram while adjusting instrumentation online and human performance issues associated with training and procedures. Implementation of the EPU requires adjustment of the direct scram from the turbine stop valve and the turbine control valve fast closure and the main steamline high flow isolation setpoints. These adjustments place the plant in a configuration that results in generation of a half scram signal and an increased potential for an unnecessary full scram of the plant. Implementation of the EPU also requires adjustment of the average power range monitor (APRM) setpoints, including the APRM simulated thermal power scram.

In addition, the licensee has identified approximately 20 instrumentation and controls and 30 operations procedures that would require revisions prior to and after the issuance of the uprate amendments if they are not issued prior to Unit 2 startup. This could result in human factor concerns associated with procedure revisions and operator training.

Therefore, exigency is appropriate in order to allow implementation of these amendments and will result in a net benefit in plant safety and reliability.

Based on the preceding discussion, the staff has determined that the licensee timely applied for the amendments that are currently needed on an expedited basis to enhance safety. Therefore, pursuant to 10 CFR 50.91(a)(6), exigent circumstances exist that warrant the issuance of these amendments before the expiration of the 30-day notice.

11.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

I. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated based upon the following discussion:

A. Evaluation of the Probability of Previously Evaluated Accidents

The proposed extended power uprate imposes only minor increases in plant operating conditions. No changes to rated core flow, rated reactor pressure, or turbine throttle pressure are required. The higher power level will result in moderate flow increases in systems associated with the turbine cycle (e.g., condensate, feedwater, and main steam). The small increase in operating temperatures for BOP [balance of plant] support systems has no significant effect on LOCA [loss-of-coolant accident] or other accident probabilities. The extended power uprate evaluations confirm the higher power level has no significant effect on flow induced erosion/corrosion. The limiting feedwater and main steam piping flow increases were evaluated and shown to be approximately proportional to the power increase. The affected systems are currently monitored by the Plant Hatch erosion/corrosion program. Continued system monitoring provides a high level of confidence in the integrity of potentially susceptible high energy piping systems.

When required, the occurrence frequency of accident precursors and transients is addressed by applying the guidance of NRC-reviewed setpoint methodology to ensure acceptable trip avoidance is provided during operational transients subsequent to implementation of extended power uprate. The setpoint evaluation confirmed Plant Hatch extended power uprate does not increase the number of challenges to the protective instrumentation.

Plant systems, components, and structures were verified as capable of performing their intended functions under increased power conditions with a few minor exceptions.

That is, some components will be modified prior to implementation of the extended power uprate program to accommodate the revised operating conditions....The Plant Hatch extended power uprate does not significantly affect the reliability of plant equipment. In cases where plant availability could be impacted by BOP equipment performance, modifications and administrative controls will be implemented to adequately compensate. No new components or system interactions that could lead to an increase in accident probability are created due to operation at 2763 MWt [megawatts thermal].

The probability of design basis accidents (DBAS) occurring is not affected by the increased power level, since the applicable criteria established for plant equipment (e.g., ANSI Standard B3 1.1 and ASME [American Society of Mechanical Engineers] Code) will still be followed when the plant is operated at the new power level. The extended power uprate analysis basis assures the limits prescribed by the Code of Federal Regulations (CFR) (e.g., LOCA PCT [peak clad temperature], SLMPCR, 10 CFR 20) will be maintained by meeting the appropriate regulatory criteria. Similarly, factors of safety specified by application of the CFR design rules were demonstrated to be maintained, as have other margin-assuring acceptance criteria used to judge the acceptability of the plant. Established reactor scram setpoints are such that there should be no increase in scram frequency due to the increased power level. No new challenges to safety-related equipment will result. Therefore, the proposed Operating License and Technical Specifications changes do not involve a significant increase in the probability of an accident previously evaluated.

B. Evaluation of the Consequences of Previously Evaluated Accidents

ECCS-LOCA Analysis

The Plant Hatch emergency core cooling system loss-of-coolant accident (ECCS-LOCA) performance analysis was performed for extended power uprate using methodology approved by the NRC for analysis required by 10 CFR 50.46. This revised analysis utilizes the same methodology (SAFER/GESTR) as the existing ECCS-LOCA analysis. ECCS requirements assumed for extended power uprate are very similar to the existing 1986 analysis. In accordance with regulatory guidance, the Plant Hatch ECCS-LOCA analysis was performed at 102% of the new RTP of 2763 MWt, or 2818 MWt. The licensing peak clad temperature remains well below the 10 CFR 50.46 required limit of 2200°F. Therefore, the analysis demonstrates Plant Hatch will continue to comply with 10 CFR 50.46 and 10 CFR 50, Appendix K at extended power uprate conditions.

Thus, the consequences of accidents are not significantly increased at the higher power level.

Abnormal Operating Transient Analysis

An evaluation of the Plant Hatch Unit I and Unit 2 Final Safety Analysis Reports (FSARs) and reload transients was performed for extended power uprate to demonstrate the proposed maximum power level will have no adverse effect on plant safety. The evaluation was performed for a power level of 2763 MWt, with the exception of certain event evaluations that were performed at 102% of 2763 MWt. The transient analysis performed to demonstrate the acceptability of Plant Hatch extended power uprate employed the same NRC-approved methods used today.

The limiting transient events at extended power uprate conditions, including events that establish the core thermal operating limits and events that bound other transient protection criteria, were evaluated. The limiting transients were benchmarked against the existing RTP [rated thermal power] level by performance of the event analysis at both the proposed power level and the current RTP level. In addition, an expanded group of transient events was evaluated to confirm these events remained less limiting than the most limiting transients. The transient events included in the expanded group were chosen based upon events demonstrated to be sensitive to initial power level. This evaluation confirmed the existing set of limiting transient events remains valid for the Plant Hatch extended power uprate. The evaluation was performed for a representative core and demonstrates the overall capability to meet all transient safety criteria. Cycle-specific analyses will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and establish cycle-specific operating limits.

The results of the limiting transients evaluation demonstrate extended power uprate can be accomplished without a significant increase in the consequences of the transients evaluated. The fuel thermal-mechanical limits at extended power uprate conditions are within the specific design criteria for the GE fuels currently loaded in the Plant Hatch cores. Also, the power-dependent and flow-dependent minimum critical power ratio (MCPR) and maximum average planar linear heat generation rate (MAPLHGR) limits utilized at Plant Hatch since the mid-1980s require only minor changes. The peak reactor pressure vessel (RPV) bottom head pressure remains within the ASME Code requirement for RPV overpressure protection. The effects of plant transients were evaluated by assessing disturbances caused by a malfunction or single failure of equipment, or operator error, consistent with the FSARs [Final Safety Analysis Reports]. Limiting transient events tend to be slightly more severe ([approximately equal to] 1%) when initiated from the new power level, assuming a 1.12 safety limit (SLMCPR) which was determined using the latest NRC-approved methods. However, for the most limiting transient, an evaluation of a representative core showed little or no change is required to the operating limit MCPR (OLMCPR) at extended power uprate and the integrity of SLMCPR is maintained. The margin of safety established by the SLMCPR is not affected and the event consequences are not significantly affected by the proposed extended power uprate to 2763 MWt. Cycle-specific analyses will continue to be performed for each fuel reload to demonstrate compliance with the applicable transient criteria and establish cycle-specific operating limits.

The transient analysis results demonstrate the Plant Hatch core thermal power output can be safely increased to 2763 MWt without significantly affecting the consequences of previously evaluated postulated transient events. The results of the extended power uprate transient evaluation are summarized as follows:

1. Events Resulting in Nuclear System Pressure Increase

   a. Main Generator Load Rejection with No Steam Bypass

      At extended power uprate conditions, the fuel transient thermal and mechanical overpower results remain below the NRC-accepted design criteria.

   b. Main Turbine Trip with No Steam Bypass

      At extended power uprate conditions, the fuel transient thermal and mechanical overpower results remain below the NRC-accepted design criteria.

   c. Main Steam Isolation Valve (MSIV) Closure

      At extended power uprate conditions, this event (with a scram initiated by the valve closure) remains nonlimiting with respect to fuel thermal limits.

   d. Pressure Regulator Failure - Closed and Slow Closure of a Single TCV [temperature control valve]

      These transients remain nonlimiting as compared with other more severe pressurization events.

2. Event Resulting in a Reactor Vessel Water Temperature Decrease

   a. Loss of Feedwater Heating

      The consequences of this event at the extended power uprate conditions remain nonlimiting with regard to the cycle OLMCPR. The results at low core flow conditions are actually slightly higher than for the high core flow condition because of increased inlet coolant subcooling into the reactor core. The calculated thermal and mechanical overpower limits at extended power uprate conditions for this event also meet fuel design criteria.

b. Inadvertent High Pressure Coolant Injection (HPCI) Actuation

For the limiting condition analyzed, both the high water level setpoint and the high RPV steam dome pressure scram setpoints are not reached. Based upon the peak average fuel surface heat flux results, the HPCI actuation event will be bounded by the limiting pressurization event with respect to delta critical power ratio ([delta]CPR) considerations. In addition, the fuel transient thermal and mechanical overpower limits remain within the allowable NRC-accepted design values.

c. Shutdown Cooling Residual Heat Removal (RHR) Malfunction

This event is not affected by extended power uprate.

3. Event Resulting in a Positive Reactivity Insertion

Rod Withdrawal Error (RWE)

The current rod block monitor (RBM) system with power-dependent setpoints was analyzed for the RWE event at extended power uprate conditions using a statistical approach consistent with NRC approved methods. The analysis concluded the transient is slightly more severe with a greater [delta]CPR from the initial most limiting CPR. However, the fuel and mechanical overpower limits remain within the NRC accepted design criteria.

4. Event Resulting in a Reactor Vessel Coolant Inventory Decrease

a. Pressure Regulator Failure to Full Open

The results of this transient for extended power uprate remain nonlimiting as compared with other more severe pressurization events.

b. Loss of Feedwater Flow

This transient event does not pose any direct threat to the fuel in terms of a power increase from the initial conditions. Water level declines rapidly and a low water level causes a reactor scram. Actuation of HPCI and reactor core isolation cooling (RCIC) terminate the event. However, the loss of feedwater flow event is included in the extended power uprate evaluation to assure sufficient water makeup capability is available to keep the core well covered when all normal feedwater is lost. A plant-specific analysis performed in support of the extended power uprate program shows a large amount of water remains above the top of the active fuel. This sequence of events does not require any new operator actions or shorter operator response times. Therefore, operator actions for the event do not significantly change for extended power uprate.

c. Inadvertent Opening of a Safety/Relief Valve (S/RV), Loss of Auxiliary Power, and Loss of One DC System

These events remain less severe at extended power uprate conditions.

5. Event Resulting in Core Coolant Flow Decrease

a. Recirculation Pump Seizure

The recirculation pump seizure transient evaluation includes the assumption the pump motor shaft of one recirculation pump stops instantaneously. As a result, core flow decreases rapidly. The heat flux decline lags core power and flow, and could result in a degradation of core heat transfer. At extended power uprate conditions, the consequences of the pump seizure event remain nonlimiting. Note the Unit 2 FSAR classifies this event as an accident due to the low probability of occurrence.

b. RPT and Recirculation Flow Control Failure Decreasing Flow

These transients remain nonlimiting at extended power uprate conditions.

6. Event Resulting in Core Coolant Flow Increase

Recirculation Flow Controller Failure Increasing Flow

The results of this transient for extended power uprate remain nonlimiting as compared with other more severe pressurization events.

7. Event Resulting in Core Coolant Temperature Increase

Failure of RHR Shutdown Cooling

This event is not significantly affected by the increase in licensed thermal power.

8. Event Resulting in Excess of Coolant Inventory

Feedwater Controller Failure - Maximum Demand

The CPR calculated for this event at extended power uprate conditions is slightly higher than the corresponding value for the current rated power. However, the trend for the feedwater controller failure - maximum demand event is consistent -with the analysis for the current rated power level. The fuel thermal margin results are within the acceptable limits for the fuel types analyzed.

DBA Challenges to Containment

The primary containment's response to the limiting DBA was evaluated at 2763 MWt, plus a 2% adder. The effect of extended power uprate on the short-term containment response (peak values), as well as the long-term containment response for containment pressure and temperature confirms the suitability of the plant for operation at the new power level. Factors of safety provided in the ASME Code are maintained, and the safety margin is not altered by uprating power to 2763 MWt.

Short-term containment response analyses were performed for the limiting DBA LOCA, a double-ended guillotine break of a recirculation suction line, to demonstrate operation at a bounding reactor power will not result in exceeding the containment design limits. This limiting DBA LOCA event results in the highest short-term containment pressures and dynamic loads. The analysis determined, at the proposed reactor power level, the maximum drywell pressure values increase only [approximately equal to] 1 psi and remain well bounded by the containment design pressure. Extended power uprate has no adverse effect on the containment structural design pressure.

Because increasing RTP increases residual heat, the containment long-term response will have slightly higher temperatures. Long-term suppression chamber temperatures remain within the design temperature of the structure; thus, ASME Code factors of safety are maintained and the safety margin is not affected. An analysis confirmed ECCS pump net positive suction head (NPSH) is not adversely affected with this temperature response, and the long-term response does not adversely affect the containment structure or the environmental qualification (EQ) of equipment located in the drywell and torus. The drywell long-term temperature response is not adversely affected for the higher reactor power; thus, the containment long-term response for extended power uprate is acceptable.

The impact of a reactor power increase on containment dynamic loads was evaluated and found to have no adverse effect for conditions that bound the proposed power level. Thus, containment dynamic loads are acceptable for operation at 2763 MWt.

The Plant Hatch extended power uprate evaluation of the primary containment response to DBAs confirmed the proposed power level does not result in a significant increase in the consequences of a postulated accident for a reactor power level [approximately equal to] 2% greater than the proposed increase to 2763 MWt.

Radiological Consequences of DBAs

For Plant Hatch extended power uprate, the radiological consequences of the limiting DBAs were reevaluated. The evaluations included the effect of the proposed power level on the radiological consequences of accidents presented in the FSARs. Reference 3 provides information on a revised radiological dose analysis for the DBA LOCA and shows doses remain within 10 CFR 100 limits at the new power level.

This DBA LOCA radiological evaluation was performed using input and evaluation techniques consistent with current regulatory guidance and appropriate plant design basis. The inputs and analysis methods are different from those utilized in the current licensing basis evaluation presented in the FSARs and the Atomic Energy Commission safety evaluation report supporting the initial plant licensing. However, the input used in the extended power uprate radiological evaluation provides a conservative assessment of the potential radiological consequences. The conclusions of these evaluations are consistent with the original licensing basis evaluations. The radiological consequences of the limiting DBA remain within 10 CFR 100 guidelines for the proposed RTP level. For the purpose of analysis, the new RTP level was increased by an additional 2% in accordance with regulatory guidance.

To demonstrate the change in consequences, the evaluation of radiological consequences using the different analysis inputs and methods was performed for the existing licensed RTP level and the proposed RTP level.

The impact of the proposed licensed power level on the fuel handling accident, control rod drop accident, and main steam line break outside primary containment was evaluated. The radiological consequences remain well below regulatory limits.

The evaluation of DBA radiological consequences confirmed extended power uprate does not result in a significant increase in consequences at a power level of 2763 MWt. The results remain below 10 CFR 100 guideline values. Therefore, the postulated radiological consequences do not represent a significant change in accident consequences and are clearly within the regulatory guidelines for the proposed power level increase.

Other Evaluations

1. Performance Improvements

   The extended power uprate safety analysis was performed taking into account the implementation of the following previously approved special operational features.

a. Single-Loop Operation (SLO)

The safety analysis for extended power conditions shows the single-loop operating mode remains valid. The current trip setpoints determined for two-loop operation (TLO) were confirmed to be acceptable for SLO, with a correction applied to account for the actual effective drive flow applied when operating with a single loop. The SLO settings were conservatively established to be consistent with the TLO settings, while ensuring the appropriate corrections are applied to the MAPLHGR and the OLCPR to account for SLO.

b. Maximum Extended Load Line Limit (MELLL)

The safety analysis for new power conditions shows the operating domain as analyzed is valid for extended power uprate conditions, even with operation permitted on a slightly higher absolute rod line.

c. Increased Core Flow (ICF)

The safety analysis for extended power uprate shows that operation at ICF conditions remains acceptable.

d. Final Feedwater Temperature Reduction (FFWTR)

The safety analysis for extended power uprate shows operation at FFWTR conditions remains acceptable.

e. Average Power Range Monitor/Rod Block Monitor Technical Specification (ARTS) Improvements

The safety analysis for extended power uprate conditions shows the ARTS improvements remain valid for the extended power uprate conditions.

2. Effect of Extended Power Uprate on Support Systems

An evaluation was performed to address the effect of the extended power uprate on accident mitigation features, structures, systems, and components within the BOP. The evaluation results are as follows:

a. Auxiliary systems, such as building heating, ventilation, and air-conditioning (HVAC) systems, reactor building closed cooling water, plant service water, spent fuel pool cooling; process auxiliaries, such as instrument air and makeup water; and the post-accident sampling system were confirmed to operate acceptably under normal and accident conditions at the proposed power level.

b. Secondary containment and standby gas treatment system were confirmed to be adequate relative to containing, processing, and controlling the release of normal and post-accident levels of radioactivity.

c. Instrumentation was reviewed and confirmed capable of performing control and monitoring functions at the proposed power level. As required, analyses were performed to determine the need for setpoint changes for various functions (e.g., APRM simulated thermal power scram setpoints). In general, setpoints are to be changed only to maintain adequate difference between plant operating parameters and trip setpoints, while ensuring safety performance is demonstrated. The revised setpoints were established using NRC-reviewed methodology as guidance.

d. Electric power systems, including the main generator and switchgear components, were verified as being capable of providing the required electrical load as a result of the increased power level. An evaluation of the auxiliary power system confirmed the system has sufficient capacity to support all required loads for safe shutdown, maintain a safe shutdown condition, and operate the required engineered safeguards equipment following postulated accidents. No safety-related electrical loads were affected which would impact the emergency diesel generators.

e. Piping systems were evaluated for the effect of operation at higher power levels, including transient loading. The evaluation confirmed piping and supports are adequate to accommodate the increased loading resulting from operation at higher power conditions.

f. The effect of the higher power conditions on a high energy line break (HELB) was evaluated. The evaluation confirmed structures, systems, and components important to safety are capable of accommodating the effects of jet impingement, blowdown forces, and the environmental effects resulting from HELB events.

g. Control room habitability was evaluated. Post-accident control room and Technical Support Center doses at 2763 MWt were confirmed to be within the guidelines of General Design Criterion 19 of 10 CFR 50, Appendix A. (See Ref. 3.)

h. The EQ of equipment important to safety was evaluated for the effect of normal and accident operating conditions at the proposed power level. The equipment remains qualified for the new conditions. The preventive maintenance program will continue to provide equipment maintenance or replacement to ensure equipment EQ at extended power uprate conditions.

3. Effect on Special Events

The consequences of special events (i.e., anticipated transient without scram (ATWS); 10 CFR 50, Appendix R; and station blackout) remain within NRC-accepted criteria at 2763 MWt. Vessel overpressure protection was analyzed assuming a closure of the MSIVs with a neutron flux scram, Although the peak reactor vessel bottom head pressure increases slightly at extended power uprate conditions, it is well within the ASME Code overpressure limit of 1375 psig. The standby liquid control (SLC) system capability analysis illustrates the plant can still achieve cold shutdown without dependence upon the control rods. Core thermal-hydraulic stability was evaluated. The new power level and modified power-to-flow map will not affect the ability to detect and suppress limit-cycle oscillations. Extended power uprate also does not adversely affect other special events, because the available equipment is not changed and the input assumptions for the evaluations are not significantly changed. Concurrent malfunctions assumed to occur during accidents were accounted for in the safety analyses for the proposed power level increase. The consequences of these equipment malfunctions do not change with the implementation of the extended power uprate program.

Conclusion

The evaluation of ECCS performance demonstrated the criteria of 10 CFR 50.46 are satisfied, thus, the margin of safety established by the criteria is maintained. The analysis demonstrated the ECCS will function with the most limiting single failure to mitigate the consequences of the accident and maintain fuel integrity. Challenges to the containment were evaluated and the integrity of the fission product barrier was confirmed. The radiological consequences of DBAs were evaluated and it was found the effect of the proposed extended power uprate on postulated radiological consequences does not result in a significant increase in accident consequences. The evaluations provide conservative results for the proposed power level of 2763 MWt and demonstrate the proposed extended power uprate does not result in a significant increase in accident consequences.

The abnormal transients were analyzed under extended power uprate conditions, and the analysis confirms the power increase to 2763 MWt has only a minor effect upon MCPR and the SLMCPR results. Thus, the margin of safety as assured by the SLMCPR is maintained. The effect of extended power uprate on the consequences of abnormal transients that result from potential component malfunctions is acceptable; thus, operation at the new power level does not result in a significant increase in transient event consequences.

The spectrum of analyzed postulated accidents and transients was investigated and determined to meet current regulatory criteria. In the area of core design, the fuel operating limits will still be met at the requested power level, and fuel reload analyses will show plant transients meet NRC-accepted criteria. The evaluation of accident consequences was performed consistent with the proposed changes to the plant Technical Specifications. Therefore, the proposed Operating License and Technical Specifications changes will not cause a significant increase in the consequences of an accident previously evaluated for Plant Hatch Unit 1 and Unit 2.

II. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated based upon the following discussion:

The BWR [boiling water reactor] configuration, operation, and event response is unchanged by the higher power level. Analyses of transient events confirm the same transients remain limiting and no transient events will result in a new sequence of events that could lead to a new accident scenario. The extended power uprate analyses confirm the accident progression is basically unchanged.

An increase in power level does not create a new fission product release path, or result in a new fission product barrier failure mode. The same fission product barriers, such as the fuel cladding, the reactor coolant pressure boundary (RCPB), and the reactor containment, remain in place. Fuel rod cladding integrity is ensured by operating within thermal, mechanical, and exposure design limits, and is demonstrated by the extended power uprate transient and accident analyses. Similarly, analysis of the RCPB and primary containment demonstrates the increased power level has no adverse effect upon these fission product barriers. The proposed Technical Specifications changes in support of extended power uprate implementation are consistent with the analyses, and assure transient and accident mitigation capability in compliance with regulatory requirements.

The effect of Plant Hatch extended power uprate on plant equipment was evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode resulting from the increased power was identified. The full spectrum of accident considerations defined in the FSARs was evaluated, and no new or different kind of accident resulting from the extended power uprate was identified. Extended power uprate analyses were performed using developed technology which was applied assuming the capability of existing plant equipment in accordance with existing regulatory criteria, including accepted codes, standards, and methods. GE has analyzed BWRs, with higher power densities and no new power-dependent accidents were identified. In addition, this uprate does not create any new sequence of events or failure modes that lead to a new type of accident.

All necessary actions will be taken prior to implementation of this program to ensure safety-related structures, systems, and components remain within their design allowable values and also ensure they can perform their intended functions under higher power conditions. The extended power uprate does not increase or create any new challenges to safety-related equipment or other equipment whose failure could cause a different kind of accident from that previously evaluated.

III. The proposed changes do not involve a significant reduction in a margin of safety based upon the following discussion:

The transient and accident analyses, as well as a majority of the plant-specific evaluations, to support the extended power uprate were performed at 2763 MWt and increased by an additional 2% in accordance with regulatory guidance, when applicable, for the evaluation of accidents and transients. The analyses demonstrate sufficient margins of safety exist. The evaluation of transient events and instrument setpoints demonstrate sufficient margin when compared to criteria establishing margins of safety for the proposed increase in power level.

The Plant Hatch extended power uprate analysis basis assures the power-dependent safety margin criteria prescribed by the CFR will be maintained by meeting the appropriate regulatory criteria. Similarly, factors of safety specified by application of the ASME Code design rules are maintained, as are other margin-assuring acceptance criteria used to judge the acceptability of the plant.

A. Fuel Thermal Limits

No change in the basic fuel design is required to achieve the extended uprate power level or to maintain the margins as discussed above. No increase in the allowable peak rod power is requested. The abnormal transients were evaluated at the higher power level for a representative core configuration. The analysis confirms the extended power uprate has no significant effect upon the OLMCPR or the SLMCPR. The fuel operating limits, such as MAPLHGR and the OLMCPR, will still be met at the new power level. The analyses confirm the acceptability of these operating limits for extended power uprate without an adverse effect upon margins to safety. Cycle specific analyses for each fuel reload will continue to be performed to demonstrate compliance with the applicable transient criteria and establish cycle-specific operating limits.

B. DBA Challenges to Fuel

Evaluation of the ECCS performance demonstrates the criteria of 10 CFR 50.46 are satisfied; thus, the margin of safety established by the criteria is maintained. This evaluation was performed at 2763 MWt, and increased by an additional 2% in accordance with regulatory guidance. The analysis demonstrates Plant Hatch will continue to comply with the guidance of 10 CFR 50.46 and the margin of safety established by the regulation will be maintained following the increase in power level.

C. DBA Challenges to Containment

The primary containment response to the limiting DBA was evaluated for extended power uprate. The effect of the increased power on the short-term containment response (peak values), as well as the long-term containment response, for containment pressure and temperature confirms the suitability of the plant for operation at the proposed power level of 2763 MWt. Factors of safety provided in the ASME Code are maintained and safety margin is not affected.

Short-term containment response analyses were performed for the limiting DBA LOCA, consisting of a double-ended guillotine break of a recirculation suction line, to demonstrate operation at the new reactor power will not result in exceeding containment design limits. The analyses determined the maximum drywell pressure increases only slightly and is bounded by the containment design pressure. Extended power uprate has no adverse effect on containment structural design pressure.

Long-term suppression chamber temperatures remain within the design temperature of the structure; thus, factors of safety provided in the ASME Code are maintained and the safety margin is not affected. Analyses confirm ECCS pump NPSH is not adversely affected with this temperature response, and the long-term response does not adversely affect the containment structure or the EQ of equipment located in the drywell and torus.

The impact of a reactor power increase on containment dynamic loads was evaluated and found to have no adverse effect for conditions that bound the proposed increase in power level. Thus, containment dynamic loads are acceptable for extended power uprate.

The Plant Hatch extended power uprate evaluation of the primary containment response to the DBA confirms the increased power level does not result in the reduction in a margin of safety.

D. DBA Radiological Consequences

The FSARs provide the radiological consequences for each DBA. The magnitude of the potential consequences is dependent upon the quantity of fission products released to the environment, the atmospheric dispersion factors, and the dose exposure pathways. For the case of extended power uprate, the atmospheric dispersion factors and the dose exposure pathways do not change. Therefore, the only factor that will influence the magnitude of the consequences is the quantity of activity released to the environment. This quantity is a product of the activity released from the core and the transport mechanisms between the core and the effluent release point.

The radiological consequences of DBAs were evaluated and it was found there is not a significant increase in consequences. The results remain below 10 CFR 100 guideline values. Therefore, the postulated radiological consequences are clearly within the regulatory guidelines, and all radiological safety margins are maintained for the proposed power level of 2763 MWt.

E. Transient Evaluations

The effect of plant transients was evaluated by assessing a number of disturbances of process variables, and malfunctions or failures of equipment consistent with the FSARs. The transient events tend to be slightly more severe ([approximately equal to] 1%) when initiated from the new power level, assuming a 1.12 SLMCPR, which was determined using the latest GE methods approved by the NRC. However, for the most limiting transient, an evaluation of a representative core shows no significant change to the OLMCPR is required for the new power level and the integrity of the SLMCPR is maintained.

Cycle-specific analyses for each fuel reload will continue to be performed to demonstrate compliance with the applicable transient criteria and establish cycle-specific operating limits.

The fuel thermal-mechanical limits at extended power uprate conditions are within the specific design criteria for the GE fuels currently loaded in the Plant Hatch cores. Also, the power-dependent and flow-dependent MCPR and MAPLHGR methods remain applicable. The peak RPV bottom head pressure remains within the ASME Code requirement for RPV overpressure protection.

The margin of safety established by the SLMCPR is not affected by the proposed power level increase to 2763 MWt.

F. Special Events

The event acceptance limits for special events remain unchanged for extended power uprate. For example, the peak RPV bottom head pressure remains below the 1375 psig ASME Code requirement for RPV overpressure protection. Acceptance limits for ATWS, Appendix R, and station blackout also remain unchanged.

G. Technical Specifications Changes

The Technical Specifications ensure the plant and system performance parameters are maintained at the values assumed in the safety analysis. The Technical Specifications (setpoints, trip settings, etc.) are selected such that adequate margin exists. For instruments that initiate protective functions (e.g., reactor protection system, ECCS, and containment isolation), proper account is taken of inaccuracies introduced by instrument drift, instrument accuracy, and calibration accuracy. The Technical Specifications address equipment availability and limit equipment out-of-service to assure the plant will have at least the complement of equipment available to deal with plant transients as that assumed in the safety analysis. The evaluations and analyses performed to demonstrate the acceptability of extended power uprate were performed using input consistent with the proposed changes to the plant Technical Specifications.

The events (i.e., transients and accidents) that form the Technical Specifications Bases were evaluated for extended power uprate conditions using input and initial conditions consistent with the proposed Technical Specifications changes. Although some changes to the Technical Specifications are required, no NRC acceptance limit is exceeded. Therefore, the margins of safety assured by safety limits and other Technical Specifications limits are maintained. The proposed changes to the Bases are consistent with the evaluations demonstrating acceptability of the new licensed power level of 2763 MWt.

Conclusion

The spectrum of postulated accidents and transients was investigated and was determined to meet the current regulatory criteria for Plant Hatch at extended power uprate conditions. In the area of core design, fuel operating limits will still be met at the new power level, and fuel reload analyses will show plant transients meet the NRC-accepted criteria as specified in the plant Technical Specifications. Challenges to fuel and ECCS performance were evaluated and shown to meet the criteria of 10 CFR 50.46 and 10 CFR 50, Appendix K. Challenges to the containment were evaluated and the integrity of the fission product barrier was confirmed. Radiological release events were evaluated and shown to meet the guidelines of 10 CFR 100. The proposed Operating License and Technical Specifications changes are consistent with the Plant Hatch extended power uprate evaluations. The evaluations demonstrate compliance with the margin-assuring acceptance criteria contained in applicable codes and regulations. Therefore, the proposed Operating License and Technical Specifications changes do not involve a significant reduction in the margin of safety.

REFERENCES

1. NRC letter from D. M. Crutchfield to G. L. Sozzi (GE), "Staff Position Concerning GE BWR Extended Power Uprate Program," TAC No. M91680, dated February 8, 1996.

2. NRC letter from K. N. Jabbour to J. T. Beckham, Jr., "Issuance of Amendments - Edwin I. Hatch Nuclear Plant Units I and 2," (TAC Nos. M91077 and M91078), dated August 31, 1995.

3. SNC letter BL-5356 from H. L. Sumner, Jr., to the NRC, "Revised Post-LOCA Doses," dated April 17, 1997.

Based upon the above considerations, the NRC staff concludes that the amendments meet the three criteria of 10 CFR 50.92. Therefore, the staff has made a final determination that the proposed amendments do not involve a significant hazards consideration.

12.0 STATE CONSULTATION

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

13.0 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 5, 1998 (63 FR 53473).

Accordingly, based on the Environmental Assessment, the Commission has determined that issuance of the amendments will not have a significant effect on the quality or the human environment.

14.0 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.

Attachments: References
Tables 1-5
Figures 1-6

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Date: October 22, 1998

REFERENCES

1. Letter from Southern Nuclear Operating Company to the NRC, "Edwin I. Hatch Nuclear Plant, Request for License Amendment, Extended Power Uprate Operation," dated August 8, 1997, and attachments.

2. Letter from Southern Nuclear Operating Company to the NRC, "Edwin I. Hatch Nuclear Plant, Request for Additional Information on Extended Power Uprate License Amendment Request," dated March 9, 1998.

3. Letter from Southern Nuclear Operating Company to the NRC, "Edwin I. Hatch Nuclear Plant, Request for Additional Information on Extended Power Uprate License Amendment Request," dated May 6, 1998.

4. Letter from Southern Nuclear Operating Company to the NRC, "Edwin I. Hatch Nuclear Plant, Request for Additional Information on Extended Power Uprate Licensse Amendment Request," dated July 6, 1998.

5. Letter from Southern Nuclear Operating Company to the NRC, "Edwin I. Hatch Nuclear Plant, Request for Additional Information on Extended Power Uprate License Amendment," dated July 31, 1998.

6. Letter from Southern Nuclear Operating Company to the NRC, "Edwin I. Hatch Nuclear Plant, Request for Additional Information, Extended Power Uprate License Amendment Request," dated September 4, 1998.

7. Letter from Southern Nuclear Operating Company to the NRC, "Edwin I. Hatch Nuclear Plant, Request for Additional Information Extended Power Uprate License Amendment Request," dated September 11, 1998.

8. Letter from Southern Nuclear Operrating Company to the NRC, "Edwin I. Hatch Nuclear Plant, Revised Post-LOCA Doses," dated April 17, 1997.

9. General Electric Licensing Topical Report NEDC-32424P, "Generic Guidelines of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR1), dated February 1995. (Proprietary information, not publicly available)

10. NRC letter from D.M. Crutchfield to G.L. Sozzi (GE), "Staff Position Concerning GE BWR Extended Power Uprate Program," dated February 8, 1996.

11. General Electric Licensing Topical Report NEDC=32523P, "Generic Evaluation of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR2), dated March 18, 1996 (Proprietary information, not publicly available) and Supplement 1, Volumes 1 and 2, dated June 1996. (Proprietary information, not publicly available).

12. NRC letter from T. H. Essig to J. F. Quirk (GE), "Staff Evaluation of General Electric Boiling Water Reactor Extended Power Uprate Generic Analyses (TAC No. M95087)," dated September 14, 1998.

13. SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," dated February 18, 1997.

14. NRC letter from Ngoc B. Le, "Safety Evaluation Related to NRC Bulletin 96-02, ' Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors' - Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TAC NOS. M95148 and M95149)," dated June 17, 1997.

15. NRC letter from K.N. Jabbour to Southern Nuclear Operating Company, "Issuance of Amendments - Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TAC NOS. M91077 and M91078)," dated August 31, 1995.

16. NRC letter from K.N. Jabbour to Georgia Power Company, "Issuance of Amendment - Edwin I. Hatch Nuclear Plant, Unit 2 (TAC NO. M87850), dated March 17, 1994.

TABLES

(Tables 1 -5)

Table 1 - Peak Fluences for Plant Hatch Unit 1 Beltline Materials at 32 EFPY

Component Heat or Heat/Lot	Surface Fluence (n/cm^2)	1/4 T Fluence (n/cm^2)	3/4 T Fluence (n/cm^2)
Plates, Lower Course:			
C4112-1, C4112-2, C4149-1	1.32×10^{18}	9.00×10^{17}	4.18×10^{17}
Plates, Lower-Int. Course:			
C4337-1, C3985-2 C4114-2	1.94×10^{18}	1.40×10^{18}	7.37×10^{17}
Welds, Lower, Longitudinal:			
13253, 1092	1.32×10^{18}	9.00×10^{17}	4.18×10^{17}
Welds, Lower-Int., Longitudinal:			
1P2809	1.94×10^{18}	1.40×10^{18}	7.37×10^{17}
1P2815	1.94×10^{18}	1.40×10^{18}	7.37×10^{17}
Welds, Circumferential:			
90099	1.32×10^{18}	9.00×10^{17}	4.18×10^{17}
33A277, 0091	1.32×10^{18}	9.00×10^{17}	4.18×10^{17}

Table 2 - Peak Fluences for Plant Hatch Unit 2 Beltline Materials at 32 EFPY

Component Heat or Heat/Lot	Surface Fluence (n/cm^2)	1/4 T Fluence (n/cm^2)	3/4 T Fluence (n/cm^2)
Plates, Lower Course:			
C8553-2, C8553-1, C8571-1	1.39×10^{18}	9.47×10^{17}	4.41×10^{17}
Plates, Lower-Int. Course:			
C8554-1, C8554-2 C8579-2	2.17×10^{18}	1.57×10^{18}	8.25×10^{17}
Welds, Lower, Longitudinal:			
10137	1.39×10^{18}	9.47×10^{17}	4.41×10^{17}
Welds, Lower-Int., Longitudinal:			
51874	2.17×10^{18}	1.57×10^{18}	8.25×10^{17}
Welds, Circumferential:			
4P6052	1.39×10^{18}	9.47×10^{17}	4.41×10^{17}

TABLE 3

Radiological Consequences of Design-Basis Accidents
(rem)

Postulated Accidents	EAB		LPZ		Control Room	
	Thyroid	WB[(1)]	Thyroid	WB	Thyroid	WB
Loss of Coolant	67	2	295	2	24	< 1
Control Rod Drop Accident	1	< 1	3	<1	3	< 1
Fuel Handling Accident	32	<1	32	< 1	12	< 1

(1) Whole Body

Table 4 Control Room Relative Concentration (X/Q) Values

	Ground Release	Stack Release
0 to 2 hours	1.26E-3	4.85E-6
2 to 8 hours	3.87E-4	1.17E-6
8 to 24 hours	4.17E-4	9.69E-7
1 to 4 days	3.56E-4	8.27E-7
4 to 30 days	2.37E-4	5.49E-7

Table 5 Site Boundary Relative Concentration (X/Q) Values (1)

	Ground Release	Stack Release
0 to 0.5 hours	1.4E-4	3.7E-5
0.5 to 2 hours	1.4E-4	1.1E-5
2 to 8 hours	7.0E-5	5.6E-6
8 to 24 hours	5.0E-5	3.8E-6
1 to 4 days	2.3E-5	1.9E-6
4 to 30 days	8.0E-6	6.4E-7

(1) From Reference 16 (minimum EAB and LPZ both equal to 1250 meters)

FIGURES

(Figures 1 - 6)

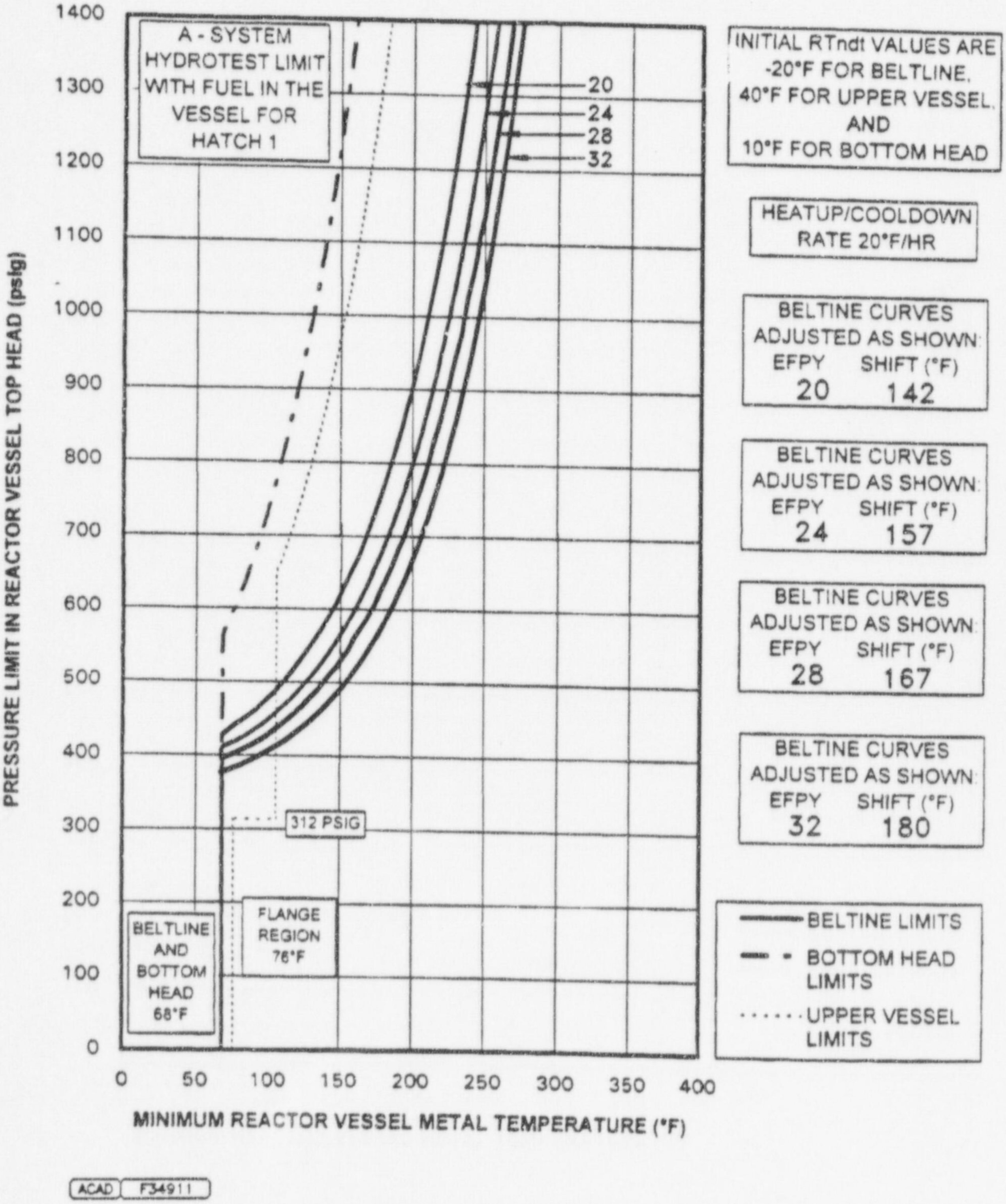

Figure 3.4.9-1 (Page 1 of 1)
Pressure/Temperature Limits for
Inservice Hydrostatic and Inservice Leakage Tests

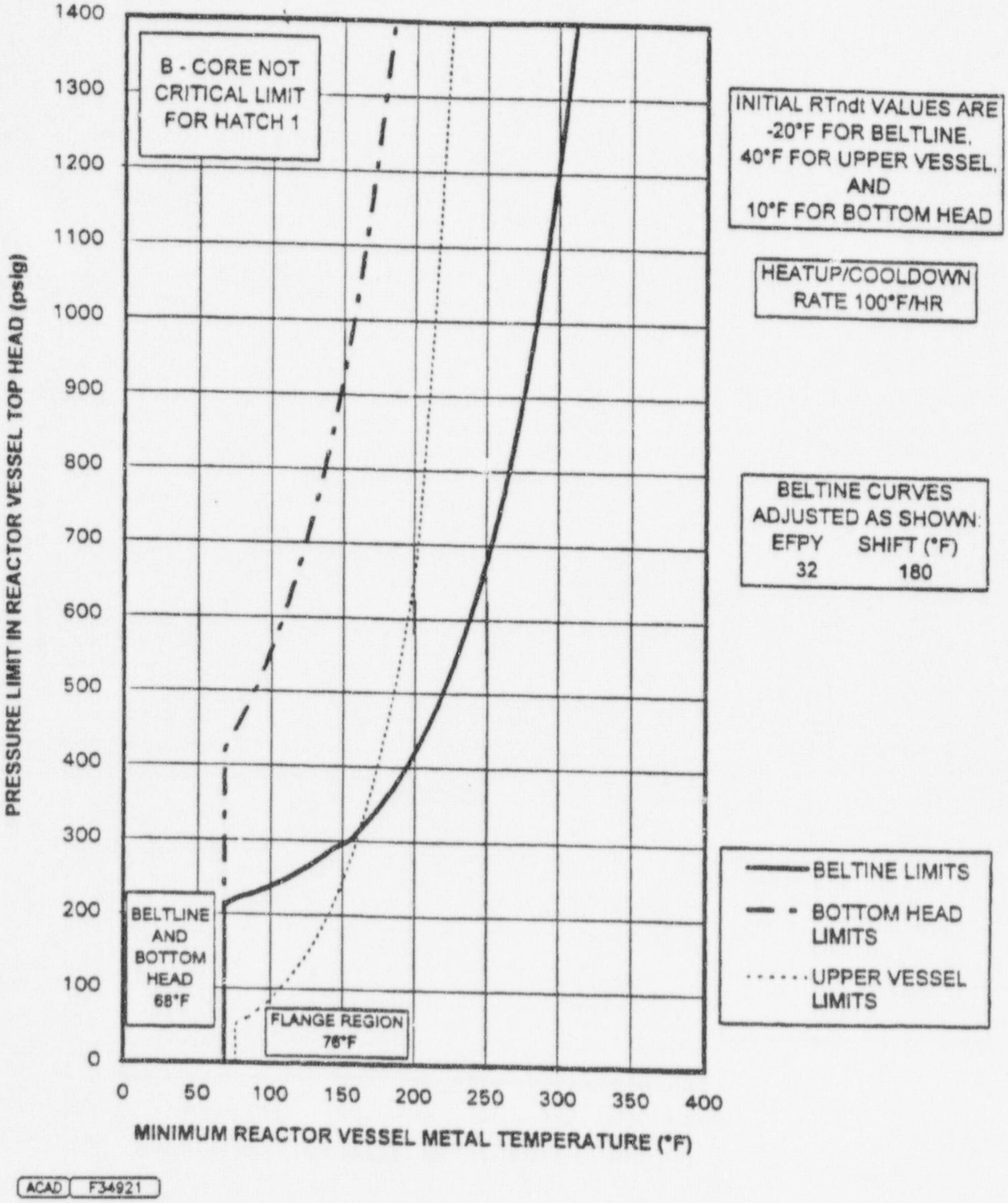

Figure 3.4.9-2 (Page 1 of 1)
Pressure/Temperature Limits for Non-Nuclear Heatup, Low Power Physics Tests, and Cooldown Following a Shutdown

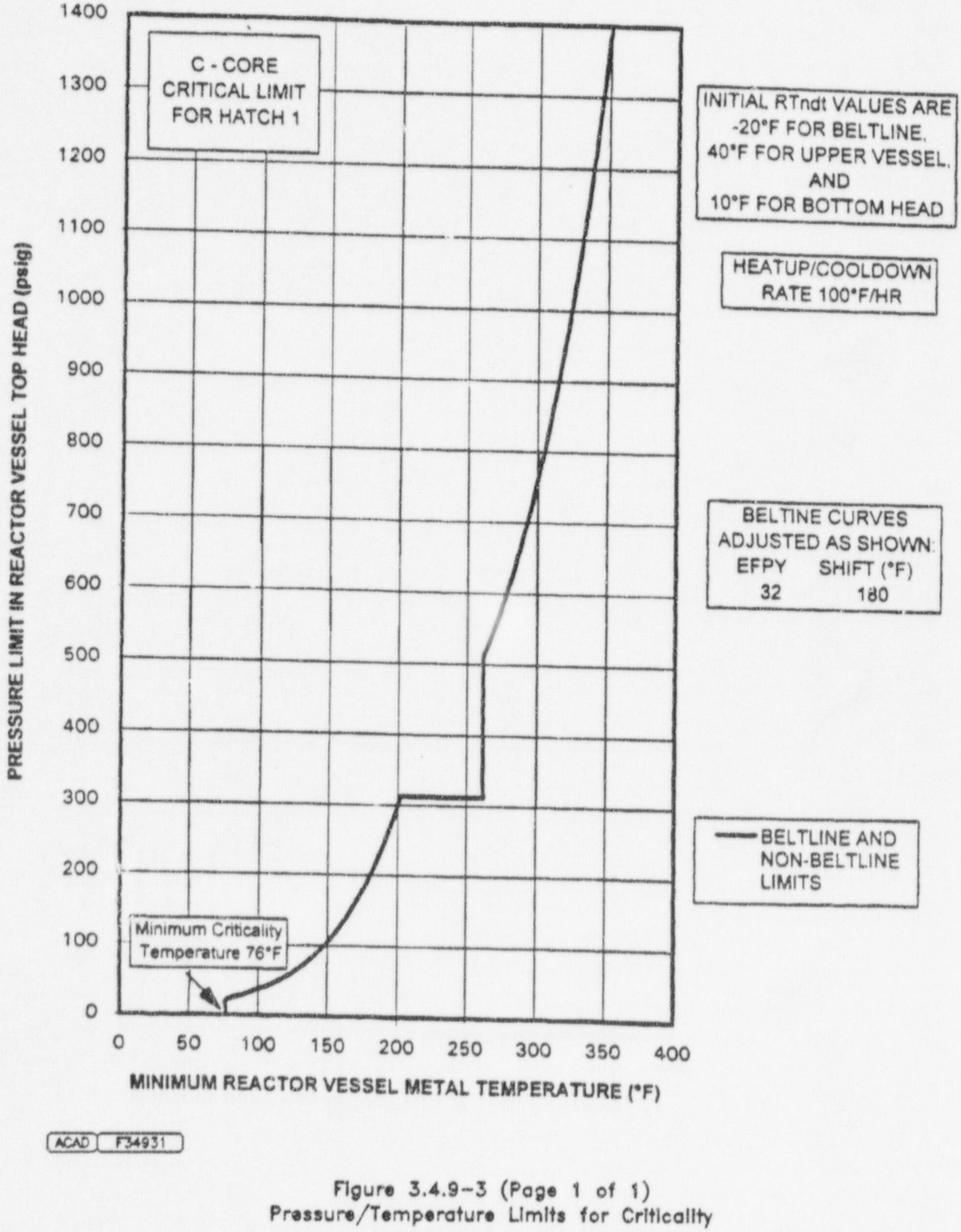

Figure 3.4.9–3 (Page 1 of 1)
Pressure/Temperature Limits for Criticality

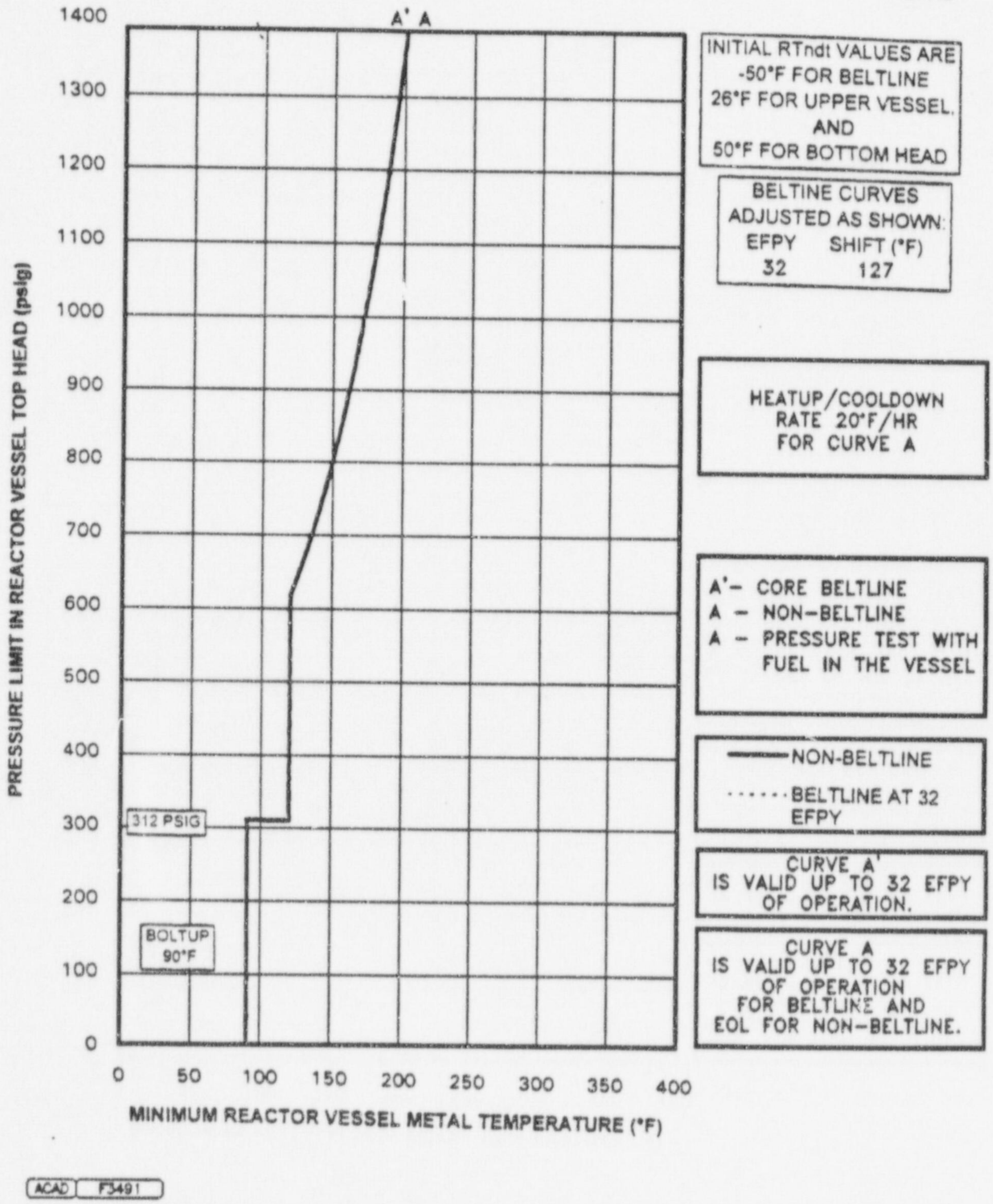

Figure 3.4.9–1 (Page 1 of 1)
Pressure/Temperature Limits for
Inservice Hydrostatic and Inservice Leakage Tests

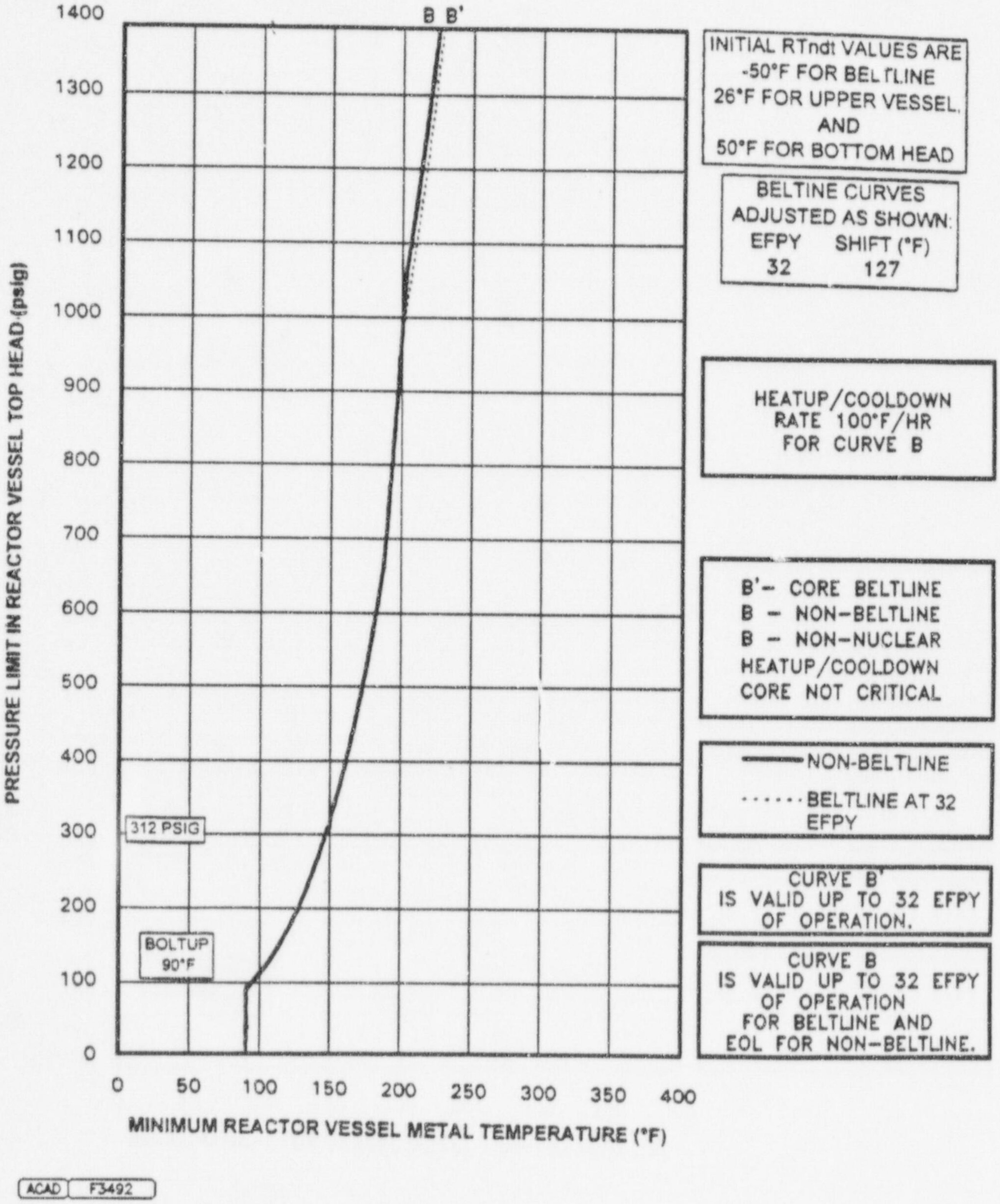

Figure 3.4.9–2 (Page 1 of 1)
Pressure/Temperature Limits for Non-Nuclear Heatup,
Low Power Physics Tests, and Cooldown Following a Shutdown

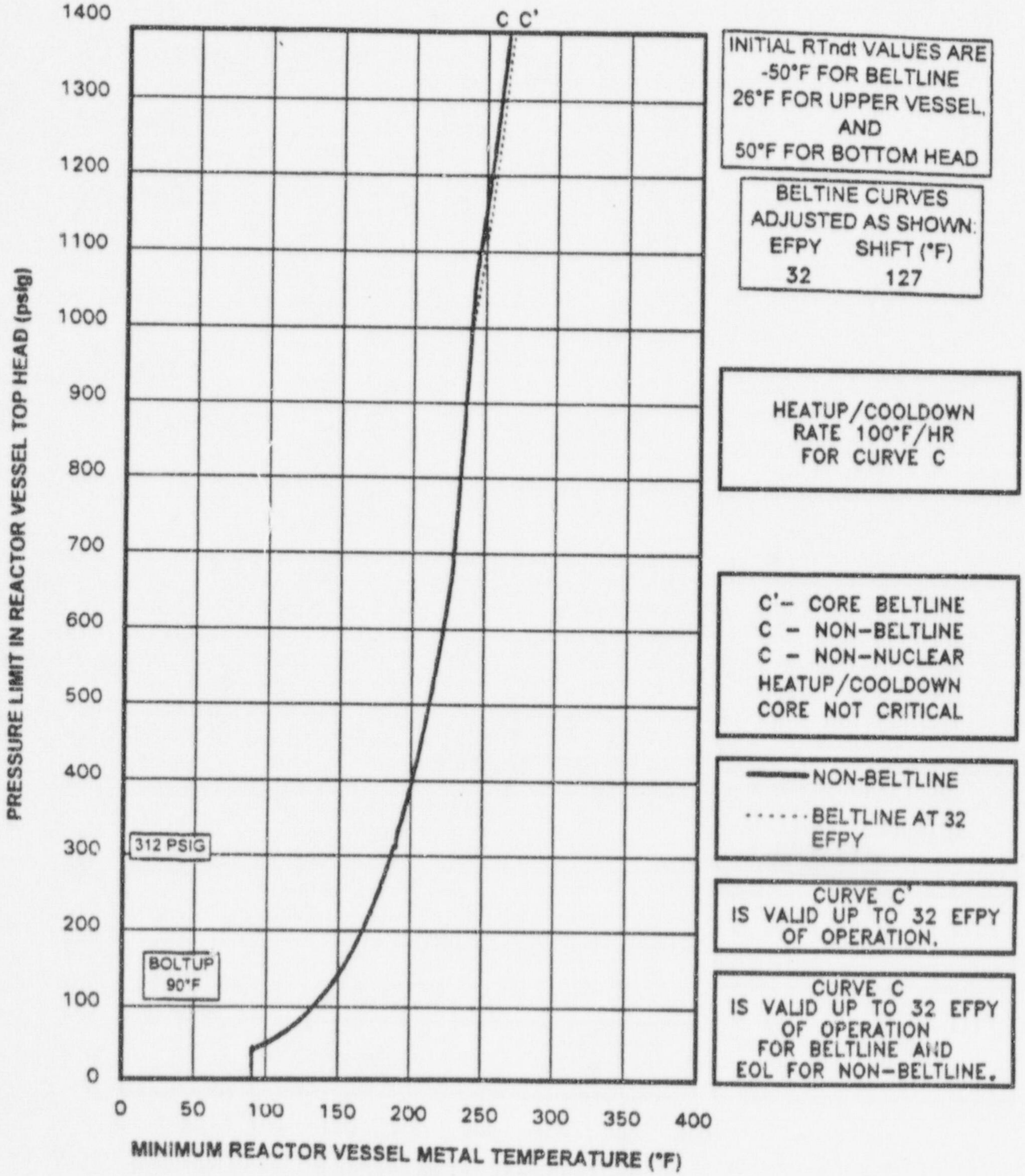

Figure 3.4.9–3 (Page 1 of 1)
Pressure/Temperature Limits for Criticality