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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 254 TO FACILITY OPERATING LICENSE NO. DPR-52

AND AMENDMENT NO. 214 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NOS. 50-260 AND 50-296

1.0 INTRODUCTION

By letter dated October 1, 1997, as supplemented October 14, 1997, March 16 and 20, April 1 and 28, May 1, 20 and 22, June 12, 17 and 26, and July 17, 24, and 31, and September 1, 1998, the Tennessee Valley Authority. (TVA or the licensee), submitted a request for changes to the Browns Ferry Nuclear Plant (BFN or the facility) Units 2 and 3, Technical Specifications (TS) to increase the maximum allowed reactor core power level for facility operation from 3293 megawatts-thermal (MWt) to 3458 MWt. The amendments also approve changes to the TS to implement uprated power operation.

The U.S. Nuclear Regulatory Commission's (NRC's) proposed action on the BFN application for an amendment was noticed on June 9, 1998 (63 FR 31533) and July 28, 1998 (63 FR 40323). The licensee provided additional details by letters dated March 20, May 22, June 12 and 17, and July 24 and 31, 1998, which did not affect the staff's proposed action described in the above-cited FR notices.

The licensee's proposal follows the generic boiling water reactor (BWR) power uprate guidelines presented in General Electric Company (GE) report NEDC 31897P-1, Generic Guidelines for General Electric Boiling Water Reactor Power Uprate, June 1991 (Reference 1). The generic analyses and evaluations in NEDC-31984P, Generic Evaluations of General Electric Boiling Water Reactor Power Uprate, July 1991; and Supplements 1 and 2 (Reference 2) are based on a slightly smaller increase (4.2 percent vs. 5.0 percent) than is requested for BFN units 2 and 3. The plant-specific analysis for BFN is presented in GE report NEDC-32751P, Power Uprate Safety Analysis Report for Browns Ferry Nuclear Plant, Units 2 and 3 (Reference 3). The licensee's request is similar to requests made by other utilities with plants similar to Browns Ferry, having BWR/4 nuclear steam supply systems and Mark 1 containment systems.

2.0 EVALUATION

The generic BWR power uprate program was created to provide a consistent means for individual licensees to recover additional generating capacity beyond their current licensed limit,

up to the reactor power level used in the original nuclear steam supply system (NSSS) design. The original licensed power level was generally based on the vendor-guaranteed power level for the reactor. Since the design power level is used in determining the specifications for major NSSS equipment, including the emergency core cooling system (ECCS), increasing the rated thermal power limits does not violate the design parameters of the NSSS equipment, nor does it significantly impact the reliability of this equipment.

BFN Units 2 and 3 are currently licensed for operation at a reactor core power level of 3293 MWt. TVA proposes to uprate the BFN units to a maximum reactor core power level of 3458 MWt. This represents approximately a 5 percent increase in the thermal power with at least a 5 percent increase in the rated steam flow. The planned approach for achieving the higher power 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 the feedwater flow, (3) no increase in the maximum core flow, (4) a small (less than 3 percent) increase in the reactor operating pressure, and (5) reactor operation primarily along extensions of pre-uprated rod/flow control lines. The operating pressure will be increased approximately 30 psi to assure satisfactory pressure control and pressure drop characteristics for the increased steam flow.

TVA has addressed the overall risk associated with the increase in rated thermal power and concluded that there is no impact on the calculated core damage frequency (CDF). Enclosure 5 to TVA's letter dated October 1, 1997, section 10.6 states that probabilistic safety assessment (PSA) evaluation of typical BWRs shows that a 5 percent power uprate has no significant impact on the CDF. TVA has reviewed its plant-specific PSA against the bases and conclusions of the above-discussed generic evaluation and confirmed that the generic conclusions are applicable to BFN.

In its review, the staff considered the recommendations from the Report of the Maine Yankee Lessons Learned Task Group, dated December 5, 1996. This report is documented in SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," dated February 18, 1997. The Task Group concluded that a power uprate review procedure should be developed in light of the Maine Yankee findings. Although a Maine Yankee lessons learned power uprate procedure has not been developed, the recommendations of the report were considered in the review of the BFN uprate. The main findings centered around the use and applicability of the code methodologies used to support the uprated power. TVA has made an effort to verify that the code inputs and assumptions are appropriate and applicable to the plant given the uprated conditions and any changes (plant modifications and procedural changes) that have occurred since initial licensing. In its May 20, 1998 letter, TVA indicated that all principal codes were used in accordance with the applicable limitations and restrictions. In its July 24, 1998, letter, TVA identified other codes (GE codes SHEX and TRACG) and other non-GE computer codes (COSMO/M and GOTHIC) that were used for the first time for BFN units and their review status i.e., generically approved. Also, in its July 24, 1998, letter, TVA confirmed that they audited GE to assure that the codes are used by GE correctly for power uprate conditions and the limitations and restrictions were followed by GE appropriately. The staff considered all of the Maine Yankee Lessons Learned recommendations and appropriately addressed them in this review.

The following sections document the staff's evaluation of the TVA's application.

3.0 REACTOR CORE AND FUEL PERFORMANCE

(1) Fuel Design and Operation

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

(2) Thermal Limits Assessment

Operating limits are established to assure regulatory and/or safety limits are not exceeded for a range of postulated events as is currently the practice. The operating and safety limits, i.e., minimum critical power ratio (MCPR) operating limit, safety limit critical power ratio, maximum average planar linear heat generation rate (MAPLHGR) and the linear heat generation (LHGR) operating limits, are cyclic dependent and as such will be established or confirmed at each reload as is described in Reference 2.

(3) Power/Flow Operating Map

The uprated power/flow operating map includes the operating domain changes for uprated power. The map includes the increased core flow (ICF) range and an uprated Maximum Extended Load Line Limit (MELLL). The maximum thermal operating power and maximum core flow correspond to the uprated power and the pre-power uprate core flow when rescaled such that uprated power is equal to 100 percent rated power. The map changes are consistent with the previously approved generic descriptions given in NEDC-31897P-A (Reference 1) and, therefore, are acceptable.

(4) Stability

The BFN units plan to implement the Option III methodology of the advanced digital power range neutron monitoring system to address the stability issue which will incorporate the power/flow map and applicable instrumentation setpoints associated with power uprate operation. By amendment Nos. 249, 253 (for BFN unit 2) and 213 (BFN Unit 3), the staff has previously approved the implementation of the advanced digital power range neutron monitoring system. In accordance with these amendments, the Oscillation Power Range Monitor (OPRM) functions will be operated in the "indicate only" mode for one fuel cycle. Following NRC staff review and approval of unit operating data, the OPRM trip function will be connected to the respective reactor protection system (RPS) channels, and OPRM-specific TS amendments will be implemented. During these test periods, the existing interim corrective actions for determining and mitigating power oscillations will remain in effect for the affected unit.

(5) Reactivity Control - Control Rod Drives (CRD) and CRD Hydraulic System

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

For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit (HCU) and the vessel bottom head is 250 psi. The CRD pumps were evaluated against this requirement and were found to have sufficient capacity. The flows required for CRD cooling and driving are assured by automatic opening of the system control valve, thus compensating for the small increase in pressure. The CRD system will continue to perform all its intended functions at uprated power, and will function adequately during insert and withdraw modes and, therefore, is acceptable.

3.2 Reactor Coolant System And Connected Systems

(1) 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 (TCVs) at the higher steam flow condition corresponding to uprated power. As a result of the power uprate, the dome pressure would increase by 30 psi and will require a change in the SRV setpoints. The licensee has proposed to change the SRV setpoints. The SRV setpoints change is appropriate and would ensure that adequate differences between operating pressure and setpoints are maintained (i.e., the "simmer margin"), and that the increase in steam dome pressure would not result in unnecessary SRV actuations. Therefore, the SRV setpoint change is acceptable.

(2) Code Overpressure Protection

The results of the overpressure protection analysis are contained in each cycle-specific reload amendment submittal. The design pressure of the reactor pressure vessel (RPV) remains at 1250 psig. The American Society of Mechanical Engineers (ASME) Code allowable peak pressure for the reactor vessel is 1375 psig (110 percent of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a main steam isolation valve (MSIV) closure with a failure of the valve position scram. The MSIV closure was analyzed by the licensee using the NRC-approved methods (ODYN), with the following

assumptions: (1) 102 percent of the uprated core power and 105 percent of core flow; (2) the maximum initial reactor dome pressure was assumed to be 1050 psig, which is higher than the nominal uprated pressure; (3) one SRV was assumed out-of-service and (4) the analysis did not take credit for externally actuated mode, via electro-pneumatic mode. The SRV opening pressures were +3 percent above the nominal setpoint for the available valves. The peak reactor pressure increases by 42 psig to 1309 psig, but remains below the ASME Code limit of 1375 psig and, therefore, the overpressure analysis is acceptable.

(3) Reactor Vessel Fracture Toughness

RPV embrittlement is caused by neutron exposure of the wall adjacent to the core (the "beltline" region). Power uprate may result in a higher neutron flux, which may increase the integrated fluence over the period of plant life. TVA evaluated the effects of increased power and pressure conditions on the RPV and internals to demonstrate compliance with 10 CFR Part 50, Appendix G. In reference 3, section 3.3.1, the licensee stated that its evaluation of the vessel in accordance with 10 CFR Part 50, Appendix G and RG 1.99, Revision 2, using the higher fluence show that:

(a) The upper shelf energy will remain greater than 50 ft-lb for the design life of the vessel and maintain the margin requirements of Appendix G.

(b) The 32 effective full power year (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. The beltline material ART will remain within the 200°F screening criterion.

(c) The pressure-temperature (P-T) curves provided in the Technical Specifications, bounded by non-beltline requirements at 12-EFPY, remain applicable to the power uprate condition. Furthermore, non-beltline requirements limit the P-T curves up to 16 EFPY for power uprate condition.

Based on its evaluation, the licensee determined that the RPV and internals will continue to meet the regulatory requirements. Since the RPV and internals continue to comply with the regulatory requirements, the staff concludes that power uprate will not adversely affect the RPV fracture toughness, and therefore, is acceptable. It is noted that these issues are applicable to the license end-of-life of the plant, and the staff may audit these issues in the future. Also, by letter dated March 3, 1998, TVA proposed a change to P-T curves which would extend their validity until 32 EFPY. This is currently under staff's review.

The effect of power uprate on the structural integrity of other reactor vessel components are addressed in section 12.0 of this safety evaluation (SE).

(4) Reactor Recirculation System

Power uprate will be accomplished by operating along extensions of rod lines on the power/flow map with no increase in maximum core flow. The cycle-specific core reload analyses will be performed with the most conservative core flow. The evaluation by the licensee of the reactor

recirculation system performance at uprated power determined that the core flow can be maintained with less than 1.3 percent increase in pump speed. The BFN units are licensed for ICF operation. TVA does not typically utilize ICF as part of the plant operational strategy and therefore has not compiled any substantial history involving operation at higher pump speeds. Therefore, TVA's experience with higher pump speed and/or vibration problems is limited. Operational limitations involving higher recirculation flow and/or vibration will be documented and resolved. Vibration monitoring is provided on Units 2 and 3 for the recirculation pump motor, pump shaft, and pump case. The licensee estimates that the required pump head and pump flow at the uprate condition will increase the power demand of the recirculation motors and the net positive suction head (NPSH), but these increases are within the capability of the equipment. The cavitation protection interlock will remain the same in absolute thermal power, since it is based on the feedwater flow rate. These interlocks are based on subcooling in the external recirculation loop and thus are a function of absolute thermal power. With power uprate, slightly more subcooling occurs in the external recirculation loop due to the higher RPV dome pressure. It would therefore be possible to lower the cavitation interlock setpoint slightly, but this change would be small and is not necessary. The licensee concluded that uprated power operation is within the capability of the recirculation system. For the reasons discussed, the staff agrees with the licensee's conclusion.

(5) Main Steam Isolation Valves (MSIVs)

The MSIVs have been evaluated by the licensee. The MSIV operating conditions under power uprate remain within the MSIV design conditions. The BFN units evaluation results are consistent with the bases and conclusions of the generic evaluation. Performance will be monitored per TS SR which would ensure original licensing basis for the MSIVs is preserved.

(6) 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 RCIC system has been evaluated by the licensee, and is consistent with the bases and conclusions of the generic evaluation. The system was found to have the capability to deliver its design rated flow at the increased reactor pressure resulting from the increase in the SRV setpoint pressure and the allowable SRV setpoint tolerance of +3 percent. The increase in reactor pressure resulting from these changes increases the maximum required pump operating head from 2800 feet to 2930 feet. To enable the RCIC system to deliver its design rate flow at the higher pump discharge head required due to power uprate, the maximum specified pump and turbine speed are increased from 4500 to 4600 rpm. Also, the surveillance test range is increased from 1010 psig and 920 psig to 1040 psig and 950 psig, consistent with the 30 psi increase to the nominal reactor operating pressure.

In response to a staff request, the licensee has indicated by letter dated May 20, 1998, that the recommendations of GE SIL No. 377 are not needed on the RCIC system on each BFN unit. This recommended modification is intended to achieve the turbine speed control/system reliability desired by SIL 377, and is consistent with the requirements in the staff SE of the generic topical report. The purpose of the modification is to mitigate the concern that a slightly higher steam pressure and flow rate at the RCIC turbine inlet will challenge the system trip

functions such as turbine overspend, high steam flow isolation, low pump suction pressure and high turbine exhaust pressure. The SIL identifies modifications primarily intended for the larger GS-2 model turbine. Although the same modification would dampen the start up transient observed in the smaller GS-1 turbine used in the BFN units, operating experience with the GS-1 indicates that it is not as susceptible to overspeed conditions during a quick start. The increase in the maximum RCIC system operating pressure resulting from power uprate is not expected to result in transient speed that requires a modification to that described in GE SIL 377. For the reasons discussed above, the staff finds that the RCIC system will deliver its design flow and, therefore, is acceptable.

(7) Residual Heat Removal System

The residual heat removal (RHR) system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal following reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The effects of power uprate on these operating modes are discussed in the following paragraphs.

(a) Shutdown Cooling Mode

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125° F in approximately 20 hours, using two RHR loops. At the uprated power level the decay heat is increased proportionally, thus slightly increasing the time required to reach the shutdown temperature to 24 hours. This increased time is judged to have an insignificant impact on plant safety. Regulatory Guide (RG) 1.139, "Guidance for Residual Heat Removal," requires demonstration of cold shutdown capability (200 degrees F reactor fluid temperature) within 36 hours. For power uprate, the licensee did not perform a plant-specific BFN evaluation for shutdown cooling based on the criteria of RG 1.139. However, as noted above, the licensee stated that the reactor can be cooled to less than 125 degrees F in 24 hours, which meets the 36-hour criterion described in RG 1.139, and therefore, the shutdown cooling operation is acceptable.

(b) Suppression Pool Cooling and Containment Spray Modes

The Suppression Pool Cooling (SPC) and Containment Spray Cooling (CSC) modes are designed to provide sufficient cooling to maintain the containment and suppression pool temperatures and pressures within design limits during normal operation and after a blowdown in the event of a design basis loss-of-coolant-accident (LOCA). This objective is met with power uprate, since the peak suppression pool temperature analysis by the licensee (described in section 4.1.1 of the licensee submittal) confirms that the pool temperature has not increased and will stay below its design limit at uprated conditions. There is no increase in the containment spray temperature. This has a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure, since these parameters reach peak values prior to actuation of the containment spray. Therefore, the staff finds that the CSC mode of operation is adequate for power uprate condition and, therefore, is acceptable.

(c) RHR System - Supplemental Fuel Pool Cooling Mode

The RHR system primarily consists of four heat exchangers and four pumps for each unit. Permanent connections with normally closed valves are provided in the shutdown cooling piping circuit for supplying cooling water to the spent fuel pool (SFP) cooling and cleanup system. In the event that the SFP heat load exceeds the heat removal capability of the SFP cooling system due to offloading the entire core, the RHR system provides supplemental cooling to the SFP. The combined heat removal capability of the SFP cooling system and the RHR system in the supplemental SFP cooling mode will maintain the SFP temperature at or below 150°F (design temperature) during a full core offload event. Heat loads on the RHR system supplemental SFP cooling mode will increase proportionally to the increase in reactor operating power level. The licensee performed evaluations and stated that the combined existing design heat removal capability of the SFP cooling system and the RHR system in the supplemental SFP cooling mode is higher than the anticipated SFP heat loads for a full core offload resulting from the proposed uprated power operations.

Based on the staff review and experience gained from its review of power uprate applications for similar BWR plants, the staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the RHR system supplemental SFP cooling mode.

(8) Reactor Water Cleanup (RWCU) System

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

4.0 <u>ENGINEERED SAFETY FEATURES</u>

4.1 Containment System Performance

The BFN Units 2 and 3, Updated Final Safety Analysis Report (UFSAR) provides the results of analyses of the containment response to various postulated accidents that constitute the design basis for the containment. Operation with power uprate of 5% from 3293 MWt to 3458 MWt would change some of the conditions and assumptions of the containment analyses. Topical Report NEDC-31897 "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," Section 5.10.2, requires the power uprate applicant to show the acceptability of the effect of the uprated power on containment capability. These evaluations will include containment pressure and temperature response, LOCA containment dynamic loads, and safety-relief valve containment dynamic loads. Appendix G of NEDC-31897 prescribes the generic approach for this evaluation and outlines the methods and scope of plant-specific containment analyses to be done in support of 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 power

uprate. The results from these analyses will be used for input to the LOCA dynamic loads evaluation. A more detailed computer model (LAMB or TRAC) may be used to determine more realistic RPV break flow rates for input to the M3CPT code. The use of the LAMB code has been previously generically reviewed by the NRC for application to LOCA analysis in accordance with 10 CFR 50, Appendix K.

Appendix G of NEDC-31897 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. These analyses can be performed using GE computer code SHEX. The analyses may use the more realistic ANS 5.1-1979 decay heat model than used for the original SAR analysis, to show compliance with temperature limits. The SHEX computer code for the calculation of suppression pool response to LOCA events has been approved on a plant-specific basis, provided that confirmatory calculations for validation of the results were included in the plant-specific request. 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).

(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 Browns Ferry UFSAR. The short-term analysis is performed to determine the peak drywell pressure during the initial blowdown of the reactor vessel inventory into containment following a large break inside the drywell (DBA LOCA), while the long-term analysis is 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-31897 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. A sensitivity analysis performed for LAMB/M3CPT for preuprate power predicted a containment pressure of 48.7 psig as compared to UFSAR value of 49.6 psig without the use of LAMB. The staff finds the use of the LAMB model detailed RPV break flow input to the M3CPT code in the containment analysis for power uprate acceptable since the difference between LAMB/M3CPT and the UFSAR values is small and the use of the LAMB model is justified generically in NEDE 20566-P-A dated September 1986.

The licensee also indicated that the SHEX code was used to model the long-term containment P-T response for power uprate. A plant-specific SHEX benchmark case using inputs consistent with the UFSAR basis using 95°F RHR service water temperature, 4500 gpm RHR service water flow rate, a RHR heat exchanger K-factor of 228 BTU/sec-°F and 6500 gpm RHR flow rate and May-Witt decay heat model was performed for BFN units as part of the power uprate analyses. The peak suppression pool temperature was predicted 176.7°F with SHEX code as compared to UFSAR value of 177°F with existing licensing basis analysis. The results of the analysis demonstrate that the peak suppression pool temperature predicted with the SHEX model are within 1°F with the existing licensing basis computer code. The shape of the long-term suppression pool temperature curve for the SHEX benchmark analysis matches well

with the corresponding curve reported in the UFSAR. Based on the comparative study results, the staff finds the use of the SHEX code for BFNP power uprate acceptable.

(a) 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 design-basis accident (DBA) LOCA. A bounding analysis was performed at 102% of the uprated power using the SHEX code and the ANS/ANSI 5.1 decay heat model (with 2σ uncertainty adder). The original analysis was performed Using May-Witt decay heat model. The staff finds the use of more realistic ANS/ANSI 5.1 decay heat model with 2σ uncertainty adder (95% confidence interval) acceptable.

The preuprate containment analysis was performed using 95°F RHR service water temperature, 4500 gpm RHR service water flow rate, an RHR heat exchanger K-factor of 228 BTU/sec-°F and 6500 gpm RHR flow rate. The uprated analysis was performed using 92°F RHR service water temperature, 4000 gpm RHR service water flow rate, an RHR heat exchanger K-factor of 223 BTU/sec-°F and the same 6500 gpm RHR flow rate. These changes are imposed in order to keep the peak suppression pool temperature after the uprate very close to the 177°F limit for the BFNP long-term torus integrity program. The 92°F RHR service water temperature is a BFNP technical specification change item. The RHR service water flow rate is conservatively assumed at a lower value of 4000 gpm to more accurately reflect system performance. The lower RHR heat exchanger K-factor is the result of the change to the RHR service water temperature and the RHR service water flow rate. The analysis shows that, using the SHEX code and ANS5.1-1979 decay heat model with 2σ uncertainty adder and the revised RHR cooling parameters as above, the SHEX predicted a peak suppression pool temperature of 175°F at the preuprate power and a 177°F at the uprate power. There will be no effect on the NPSH requirements of the ECCS pumps as the peak suppression pool temperature of 177°F remains unchanged from the current UFSAR value and is below the wetwell shell design temperature of 281°F.

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

(2) Local Suppression Pool Temperature with Main Steam Relief Valve (MSRV) Discharge

A local pool temperature for MSRV 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 MSRV discharge lines is justified in GE report NEDO-30832, "Elimination of Limit on Local Suppression Pool Temperature for SRV Discharge with Quenchers." The licensee indicated that since both units of Browns Ferry Nuclear Plant has quenchers above the RHR suction elevation, no evaluation of this limit is necessary. Based on the above review, the staff concludes that the local pool temperature limit will remain acceptable after the power uprate.

(b) Containment Gas Temperature Response

The licensee indicated that the containment gas temperature response analyses were performed at 102% of power uprate to cover the blowdown period for DBA-LOCA during which the maximum drywell airspace temperature occurs, using the Mark I containment long-term program (LTP) methodology. The power upraté analysis predicted a peak drywell airspace temperature of 297°F. The calculated peak drywell gas temperature exceeds the drywell shell design temperature of 281°F, but only at the beginning of the accident for a short period of approximately 11 seconds and does not present a threat to the drywell shell structure. The licensee also indicated that the small main steam line break (MSLB) analysis used in the equipment qualification (EQ) evaluations calculated the peak drywell airspace temperature of 336°F at the uprate condition. The total time duration for which the drywell airspace temperature exceeds the containment structural design temperature limit of 281°F is approximately 12 minutes. This temperature is not considered to present a threat to drywell shell structure, due to the short duration of the increase relative to the time required for drywell shell heatup. The calculated peak drywell shell temperature after uprate remains at 277°F and did not exceed the 281°F drywell shell design temperature limit.

The wetwell gas space peak temperature was calculated assuming thermal equilibrium between the pool and the wetwell gas space. The uprate containment analysis has calculated that the peak bulk suppression pool temperature will be 177°F after the DBA-LOCA. Due to thermal equilibrium, the maximum wetwell pool and gas space temperature will also be 177°F and, therefore, will remain below the suppression shell design temperature of 281°F.

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

(c) Short-Term Containment Pressure Response

The licensee indicated that the short-term containment response analyses were performed for the limiting DBA LOCA, which assumes 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 analysis covers 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% of power uprate level, using methods accepted during the Mark I Containment LTP. Break flow was calculated using a more detailed RPV model. The power uprate analyses predicted a maximum containment drywell pressure of 50.6 psig as compared to the preuprate UFSAR value of 49,6 psig, which remains below the BFNP containment design pressure of 56 psig. Based on its review, the staff concludes that the containment pressure response following a postulated LOCA will remain acceptable after the power uprate. The licensee will update the UFSAR to reflect the revised containment drywell pressure due to power uprate condition pursuant to 10 CFR 50.71e.

4.2 Containment Dynamic Loads

(1) LOCA Containment Dynamic Loads

Generic Guidelines in NEDC-31897 specify that the power uprate applicant determine if the containment pressure, suppression pool temperature and vent flow conditions calculated with the M3CPT code for the power uprate are bounded by the analytical or experimental conditions on which the previously analyzed LOCA dynamic loads are based. If the new conditions are within the range of conditions used to define these loads, then LOCA dynamic loads are not affected by the power uprate and thus do not require further analysis.

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

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

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

(2) Main Steam Relief Valve (MSRV) Containment Dynamic Loads

The MSRV 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 MSRV opening setpoint pressure, the initial water leg height in the SRVDL, SRVDL geometry, and suppression pool geometry. Of these parameters only the MSRV setpoint is affected by power uprate. NEDC-31897 states that if the SRV setpoints are increased, the power uprate applicant will attempt to show that the SRV design loads have sufficient margin to accommodate the higher setpoints.

The licensee indicated that the analytical limits for setpoints with power uprate are being increased by 30 psi (approximately 3%). The increased MSRV loads resulting from this increase in the setpoint pressure were compared with plant unique design limits calculated during the MARK I Containment LTP. The comparison shows that there is sufficient conservatism in the preuprate containment MSRV load definition to accommodate the increased MSRV loads due to power uprate.

Subsequent actuation loads may be affected by changes in the MSRV discharge line water level in addition to the increase in the loads due to the pressure setpoint change. For subsequent actuations (second pops), the only additional parametric change with power uprate is the time between SRV actuations. A higher water level at the time of second pop will result in higher SRV loads. The licensee stated that the effect of power uprate on the SRV discharge line was evaluated. The increased MSRV loads resulting from subsequent actuations were compared with plant unique design limits calculated during the MARK I Containment LTP. The comparison also shows there is sufficient conservatism in preuprate containment MSRV load definition to accommodate the increased MSRV loads due to subsequent actuations. Therefore, there will be no impact of power uprate on the MSRV subsequent actuation loads.

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

(3) Subcompartment Pressurization

The licensee indicated that the pressure loads on the sacrificial shield wall due to a postulated pipe break in the annulus region between the shield wall and the reactor vessel will increase slightly at uprated power. The biological shield wall designs remain adequate because the original analyzed loads were based on conservative assumptions which provide sufficient margin to accommodate the mass and energy releases at the power uprate conditions. Based on its review, the staff concludes that the subcompartment pressurization effects will remain acceptable after the power uprate.

4.3 Containment Isolation

The licensee indicated that the system designs for containment isolation logic are not affected by the power uprate. The capability of the actuation devices and air-operated valves (AOV's) to perform containment isolation function will be maintained at the uprated normal and post-accident conditions. Motor-operated valves (MOVs) used as containment isolation valves will be reevaluated to ensure compliance with Generic Letter (GL) 89-10 and 10 CFR 50.49, at the uprated and post-accident conditions. Based on above review, the staff finds that the operation of the plant at the uprated power level will not impact the containment isolation system.

4.4 Combustible Gas Control In Containment

The licensee indicated that the post-accident hydrogen and oxygen generation rates will increase in proportion to the power level. The combustibility of the post-LOCA containment atmosphere is controlled by adding nitrogen into the containment and bleeding off hydrogen and oxygen through the Standby Gas Treatment System. Sufficient capacity exists in the Containment Atmosphere Dilution (CAD) system to accommodate the slightly increased oxygen and hydrogen production due to power uprate. Initiation of the system is based upon the buildup of hydrogen and oxygen (volume fractions) in the containment following a LOCA. The original design-basis evaluations indicate the CAD initiation will be required between 1 and 2 days after the LOCA based on the suppression chamber oxygen level. Based on a 5% increase in oxygen generation rate due to power uprate, CAD initiation will be required 1 to 2

hours earlier. This slight reduction in time is not critical to manual operator action and will not result in system operation outside established design or operation restraints.

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

4.5 Emergency Core Cooling Systems (ECCSs)

The effect of power uprate and the increase in RPV dome pressure on each ECCS is addressed below. The effect of suppression pool temperature on the NPSH of the RHR pumps during the SPC and CSC modes is also discussed in section 4.1 of this SE. The review shows that there is adequate NPSH margin for the RHR and core spray (CS) pumps. Since the peak temperature is essentially the same as preuprate conditions (175°F at the preuprate power and 177°F at the uprate power), the staff concludes that the NPSH available at peak temperature conditions is not adversely affected and uprate will not affect compliance to the ECCS pump NPSH requirements.

(1) High-Pressure Core Injection System (HPCI)

The HPCI system has been evaluated by the licensee, and is consistent with the bases and conclusions of the generic evaluation. The licensee has indicated that it has implemented the guidance contained in GE SIL 480 on the HPCI system for each unit. The licensee stated that the HPCI is capable of delivering its design flow at the uprate conditions. The increase in reactor pressure resulting from the power uprate increases the maximum required pump set operating head from 2800 feet to 2930 feet. To enable the HPCI system to deliver its design rate flow at the higher pump set discharge head required due to power uprate, the maximum pump and turbine rated speed are increased from 4000 to 4100 rpm. The surveillance test range is increased from 1010 psig and 920 psig to 1040 psig and 950 psig, consistent with the 30 psi increase to the nominal reactor operating pressure. In its May 20, 1998 letter, the licensee stated that HPCI system reliability will be monitored in accordance with the criteria developed to comply with the Maintenance Rule. The staff finds that the HPCI modification will deliver its design flow for power uprate condition and the HPCI system is acceptable.

(2) Low-Pressure Core Injection System (LPCI mode of RHR)

The hardware for the low-pressure portions of the RHR are not affected by power uprate. The upper limit of the low pressure ECCS injection setpoints will not be changed for power uprate; therefore, the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low-pressure ECCS will not be increased. In addition, the RHR system shutdown cooling mode flow rates and operating pressures will not be increased. Therefore, since the system does not experience different operating conditions due to power uprate, there is no impact due to power uprate. The licensee stated that the BFN units are bounded by the generic analyses presented in section 4.1 of Reference 2 and, therefore, the staff finds the LPCI mode of RHR operation to be acceptable.

(3) Core Spray (CS) System

The hardware for the low pressure CS is not affected by the power uprate. The upper limit of the low- pressure ECCS injection setpoints will not be changed for power uprate; therefore, the low-pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. These systems do not experience different operating conditions due to power uprate, and therefore, these systems are not impacted by power uprate. Also, the impact of power uprate on the long-term response to a LOCA will continue to be bounded by the short-term response. The licensee stated that the BFN units are bounded by the generic analyses presented in section 4.1 of Reference 2. The staff finds the CS system to be unaffected by the power uprate and acceptable.

(4) Automatic Depressurization Systems (ADSs)

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

4.6 ECCS Performance Evaluation

The ECCS are designed to provide protection against hypothetical LOCAs caused by ruptures in the primary systems piping. The ECCS performance under all LOCA conditions and their analysis models must satisfy 10 CFR 50.46 and 10 CFR Appendix K requirements. The fuel, used in BFN Units, was analyzed by the licensee with the NRC-approved methods (SAFER/GESTR). The results of the base ECCS-LOCA analysis using NRC-approved methods is presented in NEDC-32484P (Reference 4), the plant-specific ECCS-LOCA results for the BFN units.

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 S/G-LOCA analysis for the BFN Units was performed by the licensee with the appropriate reload fuel in accordance with NRC requirements to demonstrate conformance with the ECCS acceptance criteria of 10 CFR 50.46 and Appendix K (Reference 4). The base S/G-LOCA and the analyses were performed at a nominal power level 3458 MWt (105 percent of the current rated power of 3293 MWt) and an Appendix K power level of 3527 MWt (102 percent of 3458 MWt) in anticipation of future power uprate. Therefore, this analysis bounds the requested power uprate of 3458 MWt. The analyses demonstrate that the limiting licensing basis peak cladding temperature (PCT) occurs for the recirculation line break DBA as documented in the base report (Reference 4), which remains applicable to the planned power uprate analysis to 3458 MWt. The analyses performed in Reference 4 are in accordance with the NRC requirements and demonstrated conformance with the ECCS acceptance criteria of 10 CFR 50.46 and, hence, it is acceptable.

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

4.7 Other Engineered Safety Feature (ESF) Systems

(1) Main Control Room Atmosphere Control System (CRACS)

The CRACS containing an emergency filtration system is designed to maintain the control room envelope at a slightly positive pressure relative to the outside atmosphere and thus minimize unfiltered inleakage of contaminated outside air into the control room following a LOCA. The licensee performed evaluations of the effects of plant operations at the proposed uprated power level on the CRACS and stated that the CRACS is not power dependent.

Based on its review and the experience gained from review of power uprate applications for similar BWR plants, the staff finds that plant operations at the proposed uprated power level do not change the design and operational aspects of the CRACS at BFN, and will have an insignificant impact on the CRACS. These issues are also discussed in section 8.0 of this SE.

(2) Emergency Cooling Water System and Auxiliary Systems

The impact of plant operations at the proposed uprated power level on safety-related and nonsafety-related water systems, other auxiliary systems and power dependent Heating, Ventilation, and Air Conditioning (HVAC) systems are addressed in section 6 of this SE.

5.0 INSTRUMENTATION AND CONTROL

The licensee stated that, based on the generic guidelines for BWR power uprates provided in GE topical reports NEDC-31897P-A and NEDC -31894P, references 1 and 2, and plant-specific methodology provided in GE topical report NEDC-32751P, reference 3, five instrument setpoint changes would have to be modified as a result of the power uprate. The staff has previously reviewed and accepted the two generic reports. These generic reports are applicable to BFN.

The staff review of the five instrumentation TS changes for the power uprate is as follows:

1. TS Table 3.3.1.1-1, Reactor Protection System Instrumentation, Function 2.b, Average Power Range Monitors, Flow Biased Simulated Thermal Power- High. Decrease the allowable value from 0.66W+71 percent rated thermal power (RTP) to 0.66W+66 percent RTP.

   The licensee stated that for the five percent power uprate, the average power range monitor (APRM) power signals will be rescaled to the uprated power level and the scram setpoints of the flow biased APRM simulated thermal power monitor will be lowered proportionally to keep these setpoints unchanged in terms of absolute power and core flow. The licensee further stated that this change will ensure that the preuprate design margins and licensing basis will be preserved for operation at the uprated power level.

2. TS Table 3.3.1.1-1, Reactor Protection System Instrumentation, Function 3, Reactor Vessel Steam Dome Pressure - High. Increase the allowable value from 1055 psig to 1090 psig.

   The licensee stated that this change is to account for the 30 psi increase in the reactor dome operating pressure. The allowable value is increased by 35 psig to place the instrument in the middle of its calibration range while preserving existing margins to reactor scram.

3. SR 3.3.4.2.3.b, ATWS-RPT Instrumentation, Reactor Steam Dome Pressure - High. Increase the allowable value from 1146.5 psig to 1175 psig.

   The licensee stated that this change is to account for the 30 psi increase in the reactor dome operating pressure. The licensee also stated that their plant-specific analysis demonstrated that this increase in the anticipated transient without scram (ATWS) recirculation pump trip (RPT) setpoint correlates with the higher analytical limit and will prevent unnecessary recirculation pump trips following pressurization transients with reactor scram. Recirculation pump operation following a scram allows for better mixing of the reactor coolant and reduces thermal stratification in the vessel.

4. SR 3.4.3.1 Main Steam Relief Valve Setpoints. Increase each setpoint by 30 psig.

   By license amendment nos. 251 and 210, the NRC previously approved these setpoint changes for BFN units 2 and 3 respectively.

5. For BFN Unit 2 only, TS Table 3.3.6.1-1, Function 5.a, RWCU System Isolation, Main Steam Valve Vault Area Temperature. Decrease the allowable temperature from 201°F to 188°F.

   The licensee stated that this change is based on their more recent RWCU pipe break analysis. The lower setpoint will detect the postulated RWCU high energy line break sooner and provide greater margin of safety by initiating more prompt RWCU isolation valve closure.

The staff review of the above instrument setpoint changes confirmed that they were established in accordance with the previously approved setpoint methodologies described in the above approved GE topical reports.

Based on the above review and justifications for the instrumentation setpoint TS changes, the staff concludes that the licensee's proposed instrument setpoint changes for power uprate are consistent with the GE power uprate methodology and plant-specific analysis previously approved by the staff, and therefore, the changes are acceptable.

6.0 AUXILIARY SYSTEMS

6.1 SFP Cooling System

The SFP cooling system, which consists of two cooling trains, is designed to maintain the SFP water temperature at or below 125°F (using both trains) during normal (partial core) refueling outages. It is noted that full core offload is not the general practice for planned refueling outages at BFN. However, TVA has previously performed full core offloads and maintains this capability. As discussed in the above section 3.7.3, supplemental fuel pool cooling is provided by the RHR system to maintain the SFP water temperature at or below 150°F in the event that a full core offload is performed.

As a result of plant operations at the proposed uprated power level, the decay heat load for any specific fuel discharge scenario will increase slightly. The licensee stated that there are administrative controls to ensure that the existing SFP cooling capacity is not exceeded during core offload.

Maintaining the SFP temperature limit at BFN is based on two primary parameters. The first is the ultimate heat sink (UHS) temperature, since the heat removal capability of the SFP cooling systems is a function of UHS temperature. The second is the core in-vessel decay time following reactor shutdown, since this determines the heat load in the SFP. Prior to planned refueling outages and full core offloads, administrative controls are in place to require analyses to be performed for determining the required core in-vessel decay time following reactor shutdown and for determining which equipment must be placed in service to maintain the pool temperature. The licensee has revised its UFSAR per 10 CFR 50.59, to reflect these administrative controls.

In addition, to enhance the assurance of BFN's ability to maintain SFP and core cooling during refueling operations, the licensee has installed a nonsafety-related alternate decay heat removal (ADHR) system. The ADHR, which has a heat removal capability higher than the combined design heat removal capabilities of the SFP cooling system, and the RHR system in the SFP cooling assist mode, is capable of cooling the entire pool and core approximately 72 hours after shutdown.

The licensee also stated that SFP heat loads and radiological consequences were evaluated for plant operations at the uprated power level. The results of the evaluation indicate that the original analyses associated with decay heat rate, time-to-boil, evaporation from boiling and the associated consequences, are still valid due to conservatism used in the original analyses.

Based on its review and the fact that the plant has administrative controls and operating procedures in place to ensure that backup cooling capability is provided for all SFP cooling scenarios, the staff finds that the design and operation of the SFP cooling systems (SFP cooling system, RHR system in the SFP cooling assist mode, and ADHR system) for the power uprate conditions meet the intent of the guidance described in the Standard Review Plan (SRP) for SFPs. The staff further finds that the SFP will be maintained below its design temperature

of 150 °F for all power uprate off-load scenarios. Therefore, operation of the SFP at power uprate conditions is acceptable.

6.2 Water Systems

The licensee stated that the environmental effects of uprate will be controlled at the same level as is presently in place. That is, the plant operation will be managed such that none of the present limits such as maximum allowed UHS temperature will be increased as a result of power uprate.

6.2.1 Raw Water Systems

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

(1) Safety-Related Loads

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

(a) Emergency Equipment Cooling Water (EECW) System

The EECW system removes heat from reactor building closed cooling water system heat exchangers, diesel generator coolers, core spray and RHR pump room coolers, RHR pump seal coolers and miscellaneous room coolers. The licensee stated that the cooling loads from these coolers with the exception of reactor building closed-cooling water (RBCCW) heat exchangers remain virtually the same as that for the current rated power level operation because the equipment performance in the areas serviced by these coolers does not change for power uprate post-LOCA conditions. The RBCCW heat loads serviced by the EECW system was evaluated and found to be acceptable. Therefore, the licensee concluded that power uprate does not require a modification of the EECW system.

Based on its review and the experience gained from review of power uprate applications for similar BWR plants, the staff finds that plant operations at the proposed uprated power level have an insignificant or no impact on the EECW system. Therefore, the staff finds that power uprate does not require a modification of the EECW system.

(b) RHR Service Water (RHRSW) System

The RHRSW system provides safety-related cooling water to the RHR system under normal or post-accident conditions. Heat loads on the RHRSW system will increase proportionally to the increase in reactor operating power level. The licensee performed evaluations and stated that the existing design heat loads for this system are higher than the anticipated equipment heat loads resulting from the proposed uprated power operations.

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

(2) Nonsafety-Related Loads

The licensee stated that the major service water heat load increases from power uprate reflect an increase in main generator losses rejected to the stator water coolers, hydrogen coolers and exciter coolers in addition to bus cooler heat loads. The increase in service water heat loads from these sources due to uprated operation is approximately proportional to the power uprate. The licensee performed evaluations which demonstrate that the service water system is adequate for power uprate conditions.

Since the nonsafety-related systems do not perform any safety-related function, the impact of the proposed uprated power operations on the designs and performances of these systems was not significant and was not reviewed.

(3) Main Condenser, Circulating Water, and Normal Heat Sink Systems

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

Since the main condenser, circulating water, and normal heat sink systems do not perform any safety-related function, the impact of the proposed uprated power operations on the designs and performances of these systems was not significant, and was not reviewed.

6.3 Reactor Building Closed-Cooling Water (RBCCW) System

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

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

6.4 Raw Cooling Water System

The heat loads on the raw cooling water (RCW) system, which are power-dependent and are increased by power uprate, are those related to the operation of the turbine-generator. The remaining RCW system heat loads are not strongly dependent upon reactor power and will not

increase significantly. The licensee performed evaluations and stated that the increase in heat loads on this system due to uprated power operations is insignificant and is within the existing design heat loads.

Since the RCW system does not perform any safety-related function, the impact of the proposed uprated power operations on the designs and performances of this system was not reviewed.

6.5 Ultimate Heat Sink

The UHS for BFN 2/3 is Wheeler Reservoir and Tennessee River. The cooling water intake temperature (upstream temperature of the river) is unaffected by uprate. The effect of power uprate on cooling water outlet temperature is an increase of less than 1°F, which is negligible.

Based on its review, the staff finds that plant operations at the proposed uprated power level will not have a significant impact on the UHS.

6.6 Standby Liquid Control System (SLCS)

The SLCS is designed to assure reactor shutdown, from full power operation to cold subcritical by mixing boron with the primary reactor coolant, in the event when no control rods can be inserted. The ability of the SLCS boron solution to achieve and maintain safe shutdown is not a direct function of core thermal power, and therefore, is not affected by power uprate. SLCS shutdown capability is re-evaluated for each reload core. The SLCS pumps are positive displacement pumps, where small pressure changes in the safety relief valve (SRV) setpoint would have no effect on the rated injection flow to the reactor. The system was found to have the capability to deliver its design rated flow resulting from the increased SRV setpoint pressure and the allowable setpoint tolerance of +3 percent. The pump surveillance test pressure is being changed from 1275 psig to 1325 psig, to account for the increase in system injection pressure at power uprate conditions.

6.7 Power Dependent Reactor Building and Plant HVAC Systems

The HVAC systems consist mainly of heating, cooling supply, exhaust and recirculation units in the reactor building, steam tunnel, diesel generator building, radwaste building and the turbine building. Power uprate results in a small increase in these system heat loads due to slightly higher process temperature and higher electrical currents in some motors and cables. The licensee stated that some areas in the reactor building, turbine building and main steam tunnel will experience slightly higher heat loads as a result of power uprate. The licensee performed evaluations and stated that, based on the small increase in overall heat load and excess design capacity, plant operations at the proposed uprated power level will have no impact on the HVAC systems for the above-cited areas.

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

6.8 Fire Protection

Fire suppression or detection is not expected to be impacted due to plant operations at the proposed uprated power level since there are no physical plant configurations or combustible load changes resulting from the uprated power operations. In addition, the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change for the uprated conditions, and the operator actions required to mitigate the consequences of a fire are not affected. The licensee performed an evaluation to demonstrate post-fire safe shutdown capability in compliance with the requirements of 10 CFR 50.48 and 10 CFR 50 Appendix R assuming power uprate conditions. The licensee concluded that plant operation at the proposed uprated power level does not affect the ability of the Appendix R systems to perform their safe shutdown function.

Based on its review of the licensee's rationale and the experience gained from review of power uprate applications for similar BWR plants, the staff concludes that the post-fire safe shutdown capability will not be affected by plant operations at the proposed uprated power level.

6.9 Systems Not Impacted By Power Uprate

The licensee identified other systems which are not affected by plant operations at the proposed uprated power level. They include:

- Auxiliary Boiler
- Condensate Makeup
- Screen Wash
- Turbine-generator Lube Oil
- Diesel Generator and its associated supporting systems
- Instrument Air
- Service Air
- Fire Detection

Based on its review and experience gained from review of power uprate applications, the staff finds that plant operations at the proposed uprated power level have no impact on these systems.

7.0 <u>POWER CONVERSION SYSTEMS</u>

7.1 Turbine-Generator

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

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

7.2 Miscellaneous Power Conversion Systems

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

Since these systems do not perform any safety-related function, the staff has not reviewed the impact of plant operations at the proposed uprated power level on the design and performance of these systems.

8.0 RADIOLOGICAL ISSUES

As discussed before, TVA will achieve the power increase by increasing core flow and by increasing the reactor vessel dome pressure to compensate for larger pressure drop through the steam lines at the 105 percent flow. Since core inventory is directly proportional to reactor power, increases in radiological releases and personnel exposures during normal operations would be limited to no greater than 105 percent. Similarly, increases in radiological consequences of design basis accidents would also be limited to 105 percent.

TVA evaluated the impact of the proposed amendment to show that the applicable regulatory acceptance criteria continue to be satisfied for the uprated power conditions. In conducting this evaluation, TVA considered the effect of the proposed higher power level on source terms, on-site and off-site doses and control room habitability during both normal operation and accident conditions.

8.1 Control Room Emergency Ventilation System (CREVS)

Since the core inventory of radionuclides (i.e., those significant in consequence assessments) is largely proportional to rated power, there could be an increase in accident radiological consequences, including increased dose to control room operators. Previously, by letter dated July 31, 1992, TVA identified certain deficiencies in the CREVS and described its corrective actions to resolve them. The staff's review of the proposed corrective actions involved three issues: atmospheric dispersion factor; LOCA release pathways; and applicability of findings on control room habitability to BFN Unit 1. Based on its review, by letter dated August 11, 1998, the staff found them to be acceptable. Therefore, the CREVS issues are not addressed here.

As part of its application for license amendment for power uprate and implementation of corrective action for CREVS issues, TVA has made the following commitments:

> TVA will perform an analysis of the design basis loss of coolant accident to confirm compliance with General Design Criteria (GDC)-9 and offsite dose limits considering main steam isolation valve leakage and emergency core cooling system leakage. The results of

this analysis will be submitted to the NRC for review and approval by March 31, 1999. Following NRC approval any required modifications will be implemented during the refueling outages scheduled for Spring 2000 for Unit 3 and Spring 2001 for Unit 2. TVA will maintain the ability to monitor radiological conditions during emergencies and administer potassium-iodide (KI) to control room operators to maintain doses within GDC-19 guidelines. This ability will be maintained until the required modifications, if any, are complete.

The above commitment has been included as a license condition in the Facility Operating License, Appendix B.

8.2 Liquid Waste Management

TVA considered changes in processing volume and radioactivity concentration. The largest source of liquid waste is associated with the backwash of the condensate demineralizers. TVA stated that the increased condensate flow would result in a decrease in the average time between backwash and precoat cycles, slightly increasing the amount of backwash liquid needing processing. TVA stated that the activated corrosion products in liquid waste were expected to increase <10 percent, but that the total volume of processed waste was not expected to increase appreciably. TVA concluded, based upon a review of plant operating effluent reports, that the requirements of Part 20 and Part 50, Appendix I, will continue to be satisfied.

Based on its review of the licensee's evaluation and experience with other similar power uprates, the staff finds that power uprate at BFN Unit Nos. 2 and 3 will not have an adverse affect on the ability of BFN to continue to meet the requirements of Part 20 and Part 50, Appendix I, for liquid effluents.

8.3 Gaseous Waste Management

The gaseous waste systems collect, control, process, store and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. At BFN, the gaseous waste management systems include the offgas system, standby gas treatment system (SGTS), and various building ventilation systems. TVA stated that these systems are designed to meet the requirements of Part 20 and Part 50, Appendix I. TVA stated that the amount of fission products released through building vents is not expected to increase significantly with an increase in power. TVA stated that the releases are administratively controlled and that the release limits are not a function of reactor power.

Based on its review of the licensee's evaluation and experience with other similar power uprates, the staff finds that power uprate at BFN Unit Nos. 2 and 3 will not have an adverse affect on the ability of BFN to continue to meet the requirements of Part 20 and Part 50, Appendix I, for gaseous effluents.

8.4 Radiation Sources in the Core and the Coolant

Radioactive materials in the reactor core are produced in direct proportion to the reactor power and the duration of irradiation. Most of the nuclides having the greatest significance with regard to dose calculations have short half lives and reach equilibrium during the operation cycle. Nuclides with longer half lives continue to accumulate with irradiation time. TVA has calculated fission product inventories based on 1400 effective full power days at a rated power of 3458 MWt. TVA states that the average irradiation time of 1400 days is appropriate for a 24-month operating cycle.

During reactor operation, the coolant passing through the core region becomes radioactive as a result of activation of impurities in the reactor water, activation of corrosion products suspended in the coolant, and release of fission products from fuel rods. Coolant and corrosion activation products in the reactor water would increase in approximate proportion to the increase in reactor power. However, the steam concentration would not change in that the increase in activity is effectively balanced by the increase in steam flow. TVA stated that while the increased feedwater flow could reduce the efficiency of the condensate demineralizers, resulting in an increase in corrosion product production, the resulting concentrations are expected to remain within the existing design basis concentrations. With regard to the release of fission products, TVA stated that although the design basis offgas activity for BFN is 0.35 Ci/s after 30 minutes of decay, observed offgas rates are well below the design basis. TVA stated that there was no expectation that the power increase would increase the rate of releases from the fuel into the reactor coolant.

Based on its review of the licensee's evaluation and experience with other similar power uprates, the staff finds that the power uprate at BFN Unit Nos. 2 and 3 will not have an adverse affect on the magnitude of radiation sources in the reactor core or reactor coolant.

8.5 Radiation Levels

TVA considered the effects of the power uprate on radiation levels in the BFN facility during normal operation as well as from post-operation and post-accident. TVA stated that the radiation levels would not increase by more than the percentage increase in reactor power, and that these increases were acceptable because many aspects of the plant were conservatively designed for higher-than-expected radiation sources. TVA also stated that increases in radiation levels would be compensated for by procedural controls, e.g., ALARA program. With regard to off-site doses during normal operations, TVA stated that the doses would not be significantly affected by operating at the uprated power level and would remain below the limits of Part 20 and Part 50, Appendix I.

Based on its review of the licensee's evaluation and experience with other similar power uprates, the staff finds that the power uprate at BFN Unit Nos. 2 and 3 will not have an adverse effect on on-site and off-site radiation levels.

8.6 Design Basis Accidents (DBAs)

TVA considered the effects of the power uprate at BFN Unit Nos. 2 and 3 on the postulated consequences of DBAs.. The magnitude of the consequences of a DBA is dependent upon the quantity of fission products released to the environment, the atmospheric dispersion factors, and the dose exposure pathways. The dose exposure pathways and the atmospheric dispersion factors are unaffected by the power uprate. (While TVA re-analyzed some dispersion factors associated with the control room intakes, the observed changes were not related to the power uprate.) The power uprate results in an increased inventory of radioactivity available in the core for release. The quantity of fission products released to the environment is dependent on the inventory available for release and the transport mechanisms between the core and the environment.

The MSLB outside of containment is based on primary coolant TS limits which are unchanged by the power uprate. The only MSLB transport mechanism potential influenced by the power uprate is the quantity of coolant mass discharged to the environment. TVA stated that the largest coolant mass release associated with an MSLB occurs at hot standby conditions which are not affected by the power uprate. For power uprate, there is a higher steam flow and, therefore, a larger quantity of steam present. TVA stated that the higher steam generation rate results in a smaller level swell resulting in less liquid mass entrained in steam. TVA concluded that the power uprate would not result in an increase in the mass loss for the MSLB.

For the LOCA, control rod drop accident (RDA) and fuel handling accidents (FHA), the only parameter dependent on the power uprate is the inventory of fission products in the core. TVA states that, for power uprates on the order of 5 percent, the increase in the quantity of fission products can be assumed to be proportional to the increase in power. TVA further stated that, since the preuprated core inventory was calculated by a method that overstated the inventory for certain classes of nuclides, the inventory for the power uprate showed increases less than 5 percent and, in some cases, slight decreases. The method used by TVA to determine power uprate core inventory is based on the industry-accepted ORIGEN-2 isotope generation and depletion computer code.

In order to determine the magnitude of the increase in postulated accident consequences due to the power uprate, TVA developed a scaling factor that was applied to the preuprate doses as a means of estimating the postuprate doses. TVA determined the scaling factor using the difference between the gamma air dose calculated using the preuprate core inventory and that calculated using the postuprate core inventory. TVA evaluated the doses to persons at the exclusion area boundary (EAB), persons within the low population zone (LPZ), and operators present within the control room.

The staff has reviewed the information provided by TVA in the amendment submittal and that provided in docketed responses to staff requests for additional information. The staff also considered information provided by TVA related to control room habitability corrective actions. Based on its review, the staff concludes that pre-uprate doses can be scaled proportionally to the amount of power increase and finds the licensee's use of ORIGEN-2 acceptable. The staff also performed confirmatory calculations to evaluate the suitability of TVA's analyses. The

parameters used by the staff are listed in Tables 1 and 2. Based on its confirmatory calculations, the staff finds reasonable assurance that the consequences of postulated DBAs will remain within 10 CFR Part 50, Appendix A, GDC-19 acceptance criteria and Part 100 limits and the BFN control room will provide adequate radiological protection such that control personnel will not receive radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body from postulated accidents occurring at Units 2 or 3. Therefore, the staff finds reasonable assurance that the radiological consequences of normal operations and anticipated accidents at the BFN Units 2 and 3 will continue to meet regulatory requirements at a rated thermal power of 3458 megawatts and, therefore, are acceptable.

Table 1

BFN Accident Analysis Parameters Used by Staff

All Accidents	
Reactor power (3458 x 1.02), MWt	3527
Core peaking factor	1.5
Number of fuel rods in core	47,368
Iodine species distribution	
Elemental	0.91
Organic	0.04
Particulate	0.05
Main condenser volume, ft^3	125,000
SGTS Flow, cfm	
Stack, Elevated	21,995
Damper bypass, ground level	5
SGTS drawdown time, sec *(assume release is ground level during drawdown)*	75
SGTS Filter Efficiency, all species, %	90
Dose conversion factors	FGR11/FGR12
Breathing rate, offsite, m^3/s	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
>24 hours	2.32E-4
Breathing rate, control room, m^3/s	3.47E-4
Control room unfiltered infiltration, cfm	3717
Control room filtered pressurization, cfm	3000
Control room volume, ft^3	210,000
Control room intake filter efficiency, all species, %	90

Control room occupancy factor	
0-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4

<u>Control Rod Drop Accident (RDA)</u>

Fraction of core Inventory in gap	
Iodine	0.1
Noble gases	0.1
Failed rods	850
Failed rod gap release fraction to vessel	
Iodine	1.0
Noble Gases	1.0
Failed rods that reach melt	0.0077
Melted fuel release fraction to vessel	
Iodine	0.5
Noble gases	1.0
Fraction of activity released to vessel that enters main condenser	
Iodine	0.1
Noble gases	1.0
Main condenser plateout fraction	
Iodine	0.9
Noble gases	0.0

<u>Case 1 Release from main condenser leakage</u>

Release rate from main condenser, %/day	1.0
Release duration, hrs	24
X/Q values	Table 2, Turbine Building

<u>Case 2 Release from main condenser via MVP to SGTS</u>

Release rate from main condenser, cfm	1850
Release duration, hrs	24
X/Q values	Table 2, Top of Stack

<u>Case3 Release via recirculation sample line to SGTS</u>

Reactor coolant volume, ft^3	26,500
Sample line release, ft^3 (lbm/hr)	220 (10651)
Flash fraction	0.36

Roof ventilator release prior to isolation, cfm	95,000
Roof ventilator isolation time, sec	7
X/Q values	Table 2, Reactor Building & Top of Stack

Loss of Coolant Accident

Containment Leakage Source

Core release fraction to CNMT	
Iodine	0.25
Noble gases	1.0
Primary CNMT volume, ft^3	285,200
Drywell	159,000
Suppression pool air space	126,200
CNMT leakrate, %/day	2.0
Secondary containment volume (50% of free volume)	1,932,000
SGTS ground level leakage (base of stack), cfm	10
Volume at base of stack (50% of free volume), ft^3	34,560
X/Q	Table 2, Top & Base of Stack

CAD System Release

Activity same as CNMT leakage case	
Flow rate, cfm	139
CAD operation, days post accident	10, 20, 29
CAD operation duration, hours	24
No mixing in RB	
X/Q	Table 2, Top & Base of Stack

MSIV Leakage*

Activity same as CNMT leakage case	
MSIV leak rate (T/S allowed corrected for temperature/pressure), ft^3/hr	32.5
Release from main condenser, %/day	0.625
Plateout fraction	0.9
X/Q values	Table 2, Turbine Building

ECCS Leakage*

Core release fraction to CNMT sump

Iodine	0.5
Noble gases	0.0
Suppression pool liquid volume, ft^3	128,700
Estimated leakage, gpm	5
Iodine Flash Fraction	0.1
Release mixes in secondary containment and released via SGTS	
X/Q	Table 2, Top & Base of Stack

**These parameters are from NRC staff analysis. As provided in the license condition, the licensee is re-performing analyses to incorporate these release pathways in the design basis.*

Fuel Handling Accident

Fuel rods damaged	125
Decay period, hrs	24
Fraction of core in gap	
I-131	0.12
Kr-85	0.3
Other iodines	0.1
Other noble gases	0.1
Pool decontamination factor	100
Roof ventilator release prior to isolation, cfm	95,000
Roof ventilator isolation time, sec	15
Mixing volume, ft3	4900
X/Q	Table 2, Top of Stack & Refuel Floor Bypass

Main Steam Line Break

Reactor coolant activity, uCi/gm dose equivalent I-131	
Normal	3.2
Spike	26
Noble gas release, uCi/sec	100,000
Mass release, lbm	
Steam	25,000
Liquid	160,000
Release flash fraction (pressure=1020 psia)	0.38
Release duration, sec	10.5
X/Q values	Table 2, Turbine Building

Table 2

BFN METEOROLOGY

Time Period	Control Room Unit 1	Control Room Unit 3	Site Boundary EAB	LPZ
Top of Stack Releases				
0-0.5 hrs	3.40E-5	3.02E-5	2.40E-5	1.30E-5
0.5-2 hrs	5.90E-15	9.64E-7	9.70E-7	8.00E-7
2-8 hrs	4.29E-15	1.89E-7		8.00E-7
8-24 hrs	3.65E-15	8.37E-8		4.00E-7
1-4 days	2.58E-15	1.43E-8		2.00E-7
4-30 days	1.57E-15	1.13E-9		6.50E-8

ref: Control Room Data from TVA ltr to NRC dtd May 1, 1998; EAB/LPZ from FSAR Table 14.6-8

Base of Stack Releases				
0-2 hrs	3.70E-3	1.2E-3	1.22E-4	5.65E-5
2-8 hrs	2.38E-3	7.91E-4		5.65E-5
8-24 hrs	1.91E-3	6.42E-4		2.24E-5
1-4 days	1.19E-3	4.09E-4		7.94E-6
4-30 days	5.97E-4	2.14E-4		1.71E-6

ref: Control Room Data from TVA ltr to NRC dtd May 1, 1998; EAB/LPZ from FSAR Table 14.6-8

Turbine Building Releases				
0-2 hrs	3.48E-4	3.48E-4	2.70E-4	1.32E-4
2-8 hrs	2.94E-4	2.92E-4		6.02E-5
8-24 hrs	2.53E-4	2.50E-4		4.07E-5
1-4 days	2.01E-4	1.97E-4		1.73E-5
4-30 days	1.44E-4	1.40E-4		5.10E-6

ref: Control Room Data from TVA ltr to NRC dtd August 10, 1994--need to be divided by two prior to use to reflect dual intakes; EAB/LPZ from TVA ltr to NRC dtd April 12, 1996

Refuel Floor Damper Bypass				
0-15 secs	1.46E-4	1.46E-4	1.22E-4	5.65E-5

ref: Control Room/EAB/LPZ from FSAR Table 14.6-8

9.0 REACTOR SAFETY PERFORMANCE FEATURES

9.1 Reactor Transients

Reload licensing analyses evaluate the limiting plant transients. Disturbances of the plant caused by a malfunction, a single failure of equipment, or personnel error are investigated according to the type of initiating event. The licensee will use its NRC-approved licensing analysis methodology to calculate the effects of the limiting reactor transients as identified in the generic guidelines. The limiting events for the BFN units were identified as those analyzed in Reference 1. The relatively small changes in rated power and maximum allowed core flow are not expected to affect the selection of limiting events. The events explicitly evaluated for the power uprate analysis are:

Loss of Feedwater Heating (LOFWH)
Feedwater Controller Failure (FWCF)
Generator Load Rejection without Bypass (GLRWOB)
Turbine Trip without Bypass (TTWOB)
Rod Withdrawal Error (RWE)
Slow Recirculation Flow Increase
Inadvertent HPCI Actuation
Loss of Feedwater Flow (LOFW)

The limiting events which establish the MCPR operating limits for the power uprate conditions are GLRWOB, TTWOB and FWCF. The analyses for the limiting transients were performed at 100 percent power. These events are analyzed with staff-approved methods ODYN code and GEMINI methodology which include allowance for core thermal power uncertainty. For events using the staff-approved REDY code, such as inadvertent HPCI actuation, the initial core thermal power is assumed as 102 percent rated, as required by REDY application. The input parameters for the transient analyses are presented in Table 9-1, and the results of the transient analyses are presented in Table 9-2 of NEDC-32751P. The analysis SRV setpoints used are shown in Table 5-1 of NEDC-32751P, with the tolerance increased to +3 percent. BFN units 2 and 3 have a virtually identical system geometry, reactor protection system configuration and mitigation functions. Additionally, both units have similar thermal-hydraulic and transient behavior characteristics. Therefore, trends with power uprate are expected to be the same for both units. The power uprate analysis used the staff-approved GEMINI methodology. Direct or statistical allowance for 2 percent power uncertainty is included in the analysis. Most of the transients events are analyzed at the full uprated power and maximum allowed core flow operating point which bounds the power/flow map shown in Figure 2-1 of NEDC-32571P.

Cycle-specific analyses will be done at each reload and the results will be part of the COLR developed by the licensee.

The safety limit minimum critical power ratio (SLMCPR) is calculated by the licensee as part of the reload licensing analyses using the NRC-approved methodology for the appropriate reload fuel. No change will be made to this methodology due to power uprate. The analysis plan

proposed by the licensee is acceptable. The licensee will submit the results of the cycle-specific analysis with each reload document.

9.2 Special Events

(1) Anticipated Transients Without Scram (ATWS)

A generic evaluation of the ATWS event is presented in section 3.7 of Supplement 1 to Reference 2. This evaluation concludes that the ATWS acceptance criteria for fuel, RPV, and the containment integrity will not be violated for power uprate if the following are met: reactor power increase is equal to or less than 5 percent; dome pressure increase is equal to or less than 40 psi; SRV opening setpoint increase is equal to or less than 80 psi; and ATWS high pressure RPT value increases equal to or less than 20 psi. The BFN power uprate meets the four criteria except that the ATWS high pressure RPT value will be increased by 30 psi. The licensee evaluated the limiting ATWS event, the MSIV closure. The RPV integrity was reanalyzed for BFN at 105 percent of pre-uprated core thermal. The results showed the peak RPV pressure to be 1439 psig, which is below the ASME code limit of 1500 psig. The peak fuel temperature is 1499 degrees F (2200 degrees limit) and the suppression pool temperature is 190 degrees F (212 degrees limit). They are acceptable.

(2) Station Blackout (SBO)

The licensee stated that the plant response and coping capabilities for an SBO event are impacted slightly by plant operations at the proposed uprated power level due to the increase in the operating temperature of the primary coolant system, decay heat, and MSRV setpoints. The licensee analyzed the impact of these increases on the condensate water requirement and the temperature heat-up in the areas which contain equipment necessary to mitigate the SBO event and concluded that no changes to the systems and equipment used to cope with to an SBO event are required.

Based on its review of the licensee's rationale and the experience gained from our review of power uprate applications for similar BWR plants, the staff finds that the impact on the coping of an SBO event due to plant operations at the proposed uprated power level will be insignificant. See section 11.0 of this SE for additional discussions relating to SBO.

10.0 <u>HIGH AND MODERATE ENERGY LINE BREAKS AND EQ OF MECHANICAL COMPONENTS</u>

(1) High Energy Line Breaks (HELBs)

The slight increase in the reactor operating P-T resulting from the plant operations at the proposed uprated power level will cause a small increase in the mass and energy release rates following an HELB outside the primary containment. This results in a small increase in the subcompartment P-T profiles. The licensee has performed an HELB analysis evaluation for systems evaluated in the FSAR which show that the affected buildings and cubicles that support the safety-related functions are designed to withstand the resulting pressure and thermal loading following a HELB.

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

(2) Moderate Energy Line Break (MELB)

The licensee determined that uprated power level operation has no impact on the MELB. Based on a review of the ECCS, the reactor core isolation cooling system, the reactor water cleanup system, and the control rod drive system, the licensee concluded that the original MELB analysis is not affected by operation at uprated power level.

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

(3) EQs of Mechanical Components

BFN was licensed prior to the establishment of NRC General Design Criterion (GDC)-4, "Environmental and Dynamic Effects Design Bases." Consequently, BFN does not have a licensing requirement to establish or maintain a formal mechanical environmental qualification program. In response to the staff's Request for Additional Information (RAI), the licensee performed an evaluation of the impact of plant operations at the proposed uprated power level on mechanical components inside and outside of the containment.

By letter dated June 26, 1998, the licensee provided the evaluation which demonstrated that there are no detrimental effects of plant operations at the proposed uprated power level on mechanical components. The licensee evaluated the conditions both internal and external to equipment.

The licensee stated that the internal conditions (temperatures, pressures, and in some cases flows) in certain systems would be affected slightly by plant operations at the proposed uprated power level. However, these changes in temperatures, pressures and flows are bounded by the original design of components.

External environmental conditions (e.g., pressure, temperature, humidity, radiation and flooding) were evaluated for normal, abnormal, and accident conditions for plant operations at the proposed uprated power level. A two-step process was followed to evaluate mechanical safety-related equipment located in harsh environments. First, a systematic room-by-room evaluation of the changes in the environments due to power uprate was performed. Secondly, for equipment in areas whose environments will be affected by power uprates, effects of environmental changes on mechanical equipment were evaluated. Based on these evaluations, the licensee concluded that there are no detrimental effects of plant operations at the proposed uprated power level on mechanical components.

Based on its review of the licensee's rationale and the experience gained from review of power uprate applications for similar BWR plants, the staff concludes that BFN 2/3 plant operation at

the proposed uprated power level will have no impact on the EQ of safety-related mechanical components inside or outside the containment and, therefore, is acceptable.

11.0 ELECTRICAL ISSUES

The staff has reviewed the main generator and its auxiliary equipment, electrical power systems, and safety-related electrical equipment environmental qualification to determine if the power uprate would have any adverse impact on the existing (onsite and offsite) power system and to see if any TS changes are necessary.

(1) Main Turbine Generator and its Auxiliary Equipment

The licensee has evaluated the power uprate of the main generator and its auxiliary equipment. The licensee reviewed the following components:

- Main Turbine Generator: For the proposed uprate of the original core thermal power, the turbine throttle pressure is required to increase from 980 psia to 1010 psia. Since the change in heat load from the current generator operating condition to the uprate condition is already factored into the design of the generator and its auxiliary equipment, and since the generators will be operating within the originally designed capability curves, no hardware changes are necessary. Thus, the licensee finds that the main generator has the capability to support the power uprate.

- Generator (Stator and Rotor) Cooling Systems: With power uprate, the generator stator bar temperature would increase without any additional cooling water flow. The licensee has evaluated the performance of the generator stator water cooler and the adequacy of its in-service stator winding assembly. The licensee has also reviewed the existing generator's rotor hydrogen coolers. Since the original generator is rated 1280 MVA at a power factor of 0.93 (i.e., 1190 MW), the proposed electrical output of 1156 MW under power uprate is still less than the originally rated generation output of 1190 MW. Therefore, the licensee finds that the generator's cooling systems are capable of reliable operation for the uprated heat load and expects no adverse impact on heat loads as a result of the power uprate condition.

- Exciter Cooling System: The licensee has reviewed the generator excitation system that consists of the exciter, voltage regulation, and rectifier, for the power uprate conditions and determined the performance of the exciter cooling system at the power uprate conditions to be adequate to handle the associated heat load.

- High-Voltage Bushings, Current Transformers, and Generator Protective Relays: For power uprate, the licensee has reviewed the in-service, high-voltage bushing and current transformers and determined that an adequate design margin exists to handle the associated heat load and increased electrical capacity. The generator protective relays have been evaluated as part of the grid stability at the new power uprate condition. The results indicate that the existing generator protective relays' performance is acceptable for power uprate.

Because the original design of the main generator has adequate capacity to handle the additional heat loads, produced by the power uprate conditions of the main generator, stator, rotor, excitation systems, bushings, and current transformers, the licensee concludes that no modifications are needed to the main generator and its auxiliary equipment. The staff finds this acceptable.

(2) Electrical Power Systems

Although no electrical hardware needs to be replaced for power uprate, it is necessary to assess the increase in the electrical power requirements for the condensate, condensate booster, and recirculation pumps in order to demonstrate that the increased load of these electrical components for power uprate has a minimal effect on BFN's electrical onsite power systems and that the increased load is well within the existing analyzed capability of the plant's onsite electrical distribution system. The licensee identified the following: (1) the condensate pump load would increase from 849 hp (preuprate) to 867 hp (power uprate), (2) the condensate booster pump load would increase from 1519 hp to 1559 hp, and (3) the recirculation pump load would increase from 8378 hp to 8678 hp. However, the licensee states that the electrical calculations for the auxiliary power system were performed using nameplate data for those affected loads (i.e., 900 hp, 1750 hp, and 9000 hp, respectively). Since the nameplate loads are higher than the operating loads at either preuprate or uprate conditions, the licensee concluded that the calculations for the preuprate are the same as for the uprate; thus, there is no need for further analysis for the proposed power uprate. On this basis, the licensee finds that the current 4-kV and the 480-V onsite power distribution system should remain adequate for the power uprate conditions.

With an increase of 57.5 MW generation to each unit, the licensee has evaluated the impact on the following components:

- Main Power Transformer (MPT): The MPT was evaluated against the maximum rated generator power production expected for the power uprate condition to establish the minimum generator power factor that could be tolerated without exceeding the transformer's 1200 MVA rating. For the proposed electrical power uprate, the licensee has determined that a generator power factor of .94 and .95 would be required for Units 2 and 3, respectively. Considering the plant house loads (through the unit station service transformers) for the onsite distribution system during normal plant operation, the MPT loadings during power uprate conditions for both units would not exceed the 1200 MVA rating of the MPT.

- Emergency Diesel Generators (EDGs): As part of the power uprate implementation, the licensee has reviewed whether all GL 89-10 MOVs would be impacted by power uprate. The results of the GL 89-10 study showed that no motor changes are required for MOVs, but the replacement of one torque switch and the reset of three other torque switches will be needed for power uprate implementation. With such minor changes for MOVs, the licensee concluded that no plant electrical equipment changes are necessary to support power uprate. Therefore, the licensee finds that there would be no impact on the EDG loading, as it would remain the same for power uprate, thus sufficient capacity exists for EDGs.

- Grid Stability: The licensee has studied generator stability associated with the increased generator capacity and has established minimum gross MVAR limits for each unit. For certain BFN 500-kV line outages, the above MVAR limitations would be incorporated into the BFN Operations Standing Orders as part of the implementation of power uprate.

- Grid Voltage and Degraded Grid Voltage Relay Setpoints: With additional generation to the BFN offsite system, the licensee has evaluated how this additional generation would affect the grid voltages and assessed whether new grid voltage would change the current grid system voltage profile, so that it affects the degraded grid voltage (DGV) relay setpoints. With preuprate switchyard voltage calculated to be 525.9 kV (nominal 500 kV) with both BFN units operating at 1100 MW, the switchyard voltage for power uprate is calculated to be 524.1 kV with both units operating at 1180 MW, which exceeds the proposed power increase of 57.5 MW. The grid system voltage profile change from preuprate to uprate conditions is 0.34%, and it has negligible impact to the plant's auxiliary power system. The licensee finds that since the grid voltage change is minimal and the methodology and software to perform the PSB-1 analysis have not changed over the years, it is not necessary to determine a new DGV relay setpoint because the DGV relay setpoints would remain the same for the power uprate.

- Isophase Bus: The isophase bus systems for BFN Units 2 and 3 are currently (preuprate conditions) carrying 32,376 amps. With the proposed uprated condition, the systems would be expected to carry 33,591 amps. Since the isophase bus cooling systems are rated for 35,270 amps, the licensee finds no problems for the isophase bus to carry the uprated 33,591 amps.

In addition, as a result of lessons learned from the Maine Yankee Independent Safety Assessment Inspection, all licensees are required to review and evaluate whether the power uprate would alter the original licensing basis for GDC-17 and the SBO requirement.

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

Plant response and coping capabilities for an SBO event could be affected by operation at the uprated power level because of the increase in the operating temperature of the primary coolant system, increase in the decay heat, and increase in the main steam safety relief valve setpoints. The licensee has re-evaluated the SBO requirement using the guidelines of NUMARC 87-00. The licensee found that: (1) the temperature increases in the control and relay rooms are not affected by power uprate; (2) the HPCI and RCIC equipment room temperature responses increase, but are within existing margins; and (3) the drywell area temperatures increase by a little amount, but the equipment necessary for event mitigation is qualified for these temperatures. Although the water requirement for the condensate increases, the current design of the condensate storage tank ensures that adequate water

volume is available. This ensures the RCIC operation throughout the coping period. On this basis, the licensee concluded that the plant will continue to meet the requirements of SBO after power uprate.

Based on the review of the licensee's evaluation, the staff finds that the electrical power system will continue to perform its intended safety-related functions for the 5% power uprate. The staff concludes that the impact of the load increase to the station auxiliary electrical distribution system is not adversely impacted by the proposed power uprate at BFN.

(3) Safety-Related Electrical Equipment Qualification

For power uprate, the licensee has evaluated the Equipment Qualification Data Packages (EQDPs) that document the qualification of safety-related electrical equipment currently installed at BFN for normal, abnormal, and accident environments. The licensee has reviewed the following areas for environmental changes on the inside and outside of the primary containment: (1) pressure, (2) temperature, (3) radiation, (4) humidity, and (5) submergence.

For inside the primary containment, short-term and long-term (100 day) containment response analyses were performed for both the design-basis LOCA and the bounding MSLB by using the GE proprietary computer codes M3CPT and SHEX. For outside primary containment, the HELB analyses were performed using the GOTHIC computer code for power uprate, and the preuprate analyses were performed using the MONSTER computer code.

The radiological analyses for power uprate are based on source-term inventories generated using the ORIGEN methodology. The preuprate radiological doses were calculated using source-term inventories based on TID-14844.

(a) Inside the Primary Containment

- Pressure: The most limiting event for drywell pressure is the design-basis LOCA for which the peak containment pressure increased from 49.6 psig to 50.6 psig with power uprate. Since the safety-related electrical equipment within scope of the BFN 10 CFR 50.49 EQ program has been evaluated for a peak containment pressure of 55 psig, the licensee found no impact on the pressure evaluations for equipment located inside the primary containment.

- Temperature: Since the normal operating temperature for the reactor vessel will increase by less than 4°F for power uprate, the licensee states that changes to normal and abnormal temperature inside the containment are expected to be small. The licensee has evaluated the normal and abnormal temperatures currently used for qualification of electrical equipment located inside the containment and determined that they bound the temperatures expected during operation at power uprate conditions.

  For the accident environment, the peak drywell temperature for a design-basis LOCA increased slightly from 295°F to 297°F; the limiting event for drywell temperature is the MSLB, for which the peak temperature increased from 322°F to 336°F. With the increased peak post-accident temperature in the drywell, the margins between the peak accident temperatures and the peak test temperatures were reduced and, in some cases, the

margin for the peak temperature was shown to be less than the 15°F suggested by IEEE Standard 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." However, although the peak containment temperature increased for the MSLB, the duration of the MSLB temperature transient is shorter and the long-term temperatures are lower than in the preuprate MSLB profile. Therefore, the power uprate MSLB profile inside the containment is actually less severe for post-accident operability than the profile used for preuprate EQ evaluations. For those cases in which the equipment test profile does not envelop the plant accident profile for the entire post-accident time, the licensee performed calculations using the Arrhenius methodology that show an increase in margin when compared to the preuprate calculations for the same equipment.

- Radiation: There is no change to accident radiation doses inside the primary containment as a result of the change from TID-14844 to the ORIGEN methodology. Although the power uprate radiological evaluations indicate that the normal radiation would increase inside the primary containment, the demonstrated (by test) dose for the safety-related electrical equipment located inside the containment was determined to be greater than the total integrated dose (normal plus accident) including margin. Therefore, the licensee found that there is sufficient margin available to accommodate the increased uprate dose.

(b) Outside the Primary Containment

- Pressure: The GOTHIC analysis results indicate that the peak pressures for HELBs outside of the primary containment are lower than the peak pressures from the preuprate MONSTER analysis. Therefore, the licensee found that there is no impact on the pressure evaluation previously performed for electrical equipment located outside of the primary containment.

- Temperature: The peak reactor building temperatures for HELB events calculated using the GOTHIC computer code were higher for certain areas than the preuprate peak temperatures calculated using the MONSTER computer code. However, the margins between the peak accident temperatures and the peak test temperatures for the safety-related electrical equipment were greater than the 15°F suggested by IEEE Standard 323-1974 with the exception of certain Limitorque motor operators in the torus room. The technical justification for accepting the less than suggested 15°F margin is based on a review of the EQ test report. The report indicates that the motor operators were tested at 250°F for more than 24 hours. Although the plant profile reaches a peak temperature of 250°F, the duration of the accident temperature transient is extremely short and the room temperature decreases to less than 180°F in 1 minute. The power uprate HELB profiles outside the primary containment are generally less severe for post-accident operability than the profiles used for preuprate EQ evaluations. Calculations performed using the Arrhenius methodology show an increase in margin when compared to the preuprate calculations for most equipment.

- Radiation: The normal operating doses for most reactor building areas were bounded by the normal radiation doses used for previous equipment qualification. In a few areas, the normal radiation doses increased as a result of operation at power uprate conditions. The only area in which the accident radiation dose increased above the dose used for previous

EQ evaluations was in the standby gas treatment building. The licensee has evaluated these radiation dose increases and found that sufficient margin exists since the total integrated dose, including margin, remains less than the demonstrated (by test) dose for the affected equipment.

- Humidity: The licensee found that there is no change to the normal, abnormal, or accident humidity values for power uprate conditions for inside and outside the primary containments.

- Submergence: The licensee found that there is no significant change to flood levels for power uprate conditions for inside and outside the primary containments.

Since the licensee's evaluation for EQDPs has not identified conditions at the uprate power level that would exceed the conditions to which the equipment was tested, the staff concludes that the safety-related electrical equipment currently installed at BFN is qualified for normal, abnormal, and accident environments at the power uprate conditions.

However, the staff noted that a degraded equivalency analysis using the Arrhenius methodology was performed to extend the test duration to the BFN-specific accident requirements. As a separate initiative outside the scope of this evaluation, the staff will continue to review the adequacy of degraded equivalency analysis using the Arrhenius methodology.

(4) Conclusion

The staff concludes that the proposed power uprate at BFN would have no adverse impact on the station's electrical system and does not require any TS change for the electrical power system.

12.0 <u>MECHANICAL ISSUES</u>

The staff's review of the licensee's submittal focused on the effects of power uprate on the structural and pressure boundary integrity of the reactor coolant piping, components, their supports, reactor vessel and internal components, the Control Rod Drive Mechanism (CRDM), certain pumps and valves, and the balance-of-plant (BOP) piping systems.

The GE generic guidelines for BWR power uprate were based on a 5% higher steam flow, an operating temperature increase of 5°F and an operating pressure increase of 40 psi or less. For BFN, the maximum reactor vessel dome pressure increases from 1005 psig to 1035 psig (increase 30 psi). The dome temperature increases from 547°F to 551°F (increase 4°F) and the steam flow rate increases approximately 6% from 13.37×10^6 lb_m/hr to 14.15×10^6 lb_m/hr for both Browns Ferry 2 and 3. The maximum core flow rate remains unchanged for the BFN power uprate conditions.

(1) RPV and Internals

The licensee evaluated the reactor vessel and internal components by considering load combinations of the plant licensing basis as specified in the BFN UFSAR. These load

combinations include reactor internal pressure difference (RIPD), LOCA, and seismic and fuel lift loads. The seismic and fuel lift loads are unaffected by the power uprate. The licensee recalculated RIPDs for the power uprate shown in Table 3-2 of Reference 3 for normal, upset and faulted conditions.

The stresses and cumulative fatigue usage factors (CUFs) for the reactor vessel components were evaluated by the licensee, in accordance with the code of record at Browns Ferry, the ASME Code Section III, Subsection NB, 1965 Edition with Summer 1965 addenda for BFN unit 2, and 1965 Edition with Summer 1966 addenda for BFN unit 3. 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 Reference 3. The CUFs for the uprated power level were calculated by using the power uprate scaling factor for limiting components such as feedwater nozzle, recirculation outlet nozzle, main closure stud and support skirt. For these limiting components, the calculated CUFs were provided in Table 3-3 of Reference 3 and the maximum stresses for the uprated power conditions were summarized in the licensee's March 16, 1998 letter, Table 1(c)-2.

The licensee stated that the CUF for the feedwater nozzle was calculated to be 0.984 excluding the inner blend radius location for which the structural integrity was demonstrated separate by the power uprate fracture mechanics calculation in accordance with NUREG-0619. The BFN triple thermal sleeve sparger design at the feedwater nozzle has seal surfaces where there could be potential leakage of the feedwater. The potential leakage can mix the feedwater with the hotter downcomer flow and cause temperature cycling which could yield a significant high cycle fatigue usage. The licensee also indicated that the current BFN NUREG-0619 program includes both the monitoring of seal leakage and the inspection of feedwater nozzle to ensure that the integrity of the nozzle will be maintained for the power uprate condition. The staff finds that the licensee's evaluation is adequate to ensure the structural integrity of feedwater nozzle at the uprated power level.

TVA letter dated March 16, 1998, Table 1-2(c) shows the stress in the support skirt in exceedance of the code limit. In its response dated July 24, 1998, to the staff's RAI, the licensee provided a detailed description and calculation results that demonstrate the reactor vessel support skirt to be acceptable using elastic-plastic analysis in accordance with the ASME Code III Section NB-3228.5. The stresses are in compliance with the Code requirements. The component meets the thermal ratcheting requirement of NB-3222.5 and the CUF for the support skirt was calculated to be less than the Code limit of 1.0. The staff finds the licensee's evaluation to be acceptable.

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

Based on its review, the staff finds that the maximum stresses and fatigue usage factors as provided by the licensee are within the code-allowable limits and concludes that the reactor

vessel and internal components will continue to maintain the structural integrity for the power uprate.

(2) Control Rod Drive Mechanism (CRDM)

The licensee evaluated the adequacy of the CRDM in accordance with the Code of record, the ASME Code Section III, 1968 Edition and Addenda to and including Summer 1970.

In a March 16, 1998 letter, 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 this component results from a maximum CRD internal hydraulic pressure of 1750 psig by an abnormal operating condition. 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.

In addition, the CRDMs have been designed for dome pressure of 1250 psig and operating temperature of 575°F which are higher than the maximum bottom head pressure and temperature of 1070 psig and 532°F, respectively, for the power uprate at BFN.

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

(3) Reactor Coolant Piping and Components

The licensee evaluated the effects of the power uprate conditions, including higher flow rate, higher temperature and higher pressure, 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 licensee indicated that the evaluation was performed using the original Code of record specified in the Browns Ferry UFSAR, 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, reactor vessel bottom head drain line, RWCU, reactor vessel head vent line, RCIC, condensate and feedwater system, CS, HPCI, RHR 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 3-4 and 3-5 of the power uprate license amendment request of Reference 3.

As summarized in Table 3-4 and 3-5 of Reference 3, 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 power uprate, and are, therefore, acceptable. For those systems whose design temperature and pressure did not envelop the uprate power conditions, the licensee reviewed the existing design basis stress analyses and compared the design margins or differences between the calculated stresses and the Code allowable limits against the percentage increases in Tables 3-4 and 3-5. The licensee concluded that the

original analyses of RCPB piping systems have sufficient design margins to accommodate the higher operating flow, pressure and temperature due to the proposed power uprate. 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, appropriate components, connections and supports in a manner similar to the evaluation of the RCPB piping systems based on increases in temperature and pressure of the design basis analysis input. These systems include lines which are affected by the power uprate, but not evaluated in section 3.5 of Reference 3, such as main steam bypass lines, the MSRV discharge, and portions of main steam and feedwater systems outside the primary containment. The licensee reviewed the maximum stress levels and fatigue analysis results and found that there are sufficient margins between actual stresses and Code-allowable limits in the original design to accommodate the specific increases in temperature, pressure and flow rate provided in Tables 3-4 and 3-5. In addition, the licensee indicated that the LOCA and MSRV discharge dynamic loads for the proposed power uprate are within those loads in the existing analysis basis. The piping systems attached to the torus shell are designed for a temperature limit of 177°F which remains the same for the power uprate. The licensee concluded that all piping is below the Code-allowable limits. The staff reviewed the licensee's submittals and finds that the licensee's evaluations are acceptable.

The licensee evaluated pipe supports including anchorages, equipment nozzles, and penetrations by comparing the increased piping interface loads on the system components due to the power uprate thermal expansion, with the margin in the original design basis calculation, and performing detailed analyses using exact load combinations at the uprated conditions. The effect of power uprate conditions on thermal and vibration displacement limits was also evaluated by the licensee for struts, springs and pipe snubbers, and found to be acceptable. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the power uprate, and no new pipe break locations were identified. The staff finds the licensee's evaluation to be acceptable.

Based on the above 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 power uprate.

(4) Equipment Seismic and Dynamic Qualification

The licensee evaluated equipment qualification for the power uprate condition. The dynamic loads such as MSRV 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 Reference 3 because the plant-specific hydrodynamic loads defined during the Mark I Containment LTP for the design-basis analysis at Browns Ferry are bounding for the power uprate.

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

a. Seismic loads are unchanged for the power uprate;
b. No new pipe break locations were identified for the power uprate;
c. MSRV and LOCA dynamic loads, and pipe-whip and jet impingement loads used in the original design basis analyses are bounding for the power uprate.

(5) Safety-Related Safety/Relief Valves and Power-Operated Valves

In its May 22, 1998, response to the staff's RAI dated April 22, 1998, the licensee indicated that the current NRC-approved MSRV setpoint tolerance (+/-3%) was used in the abnormal transient analyses for the power uprate condition. The MSRV setpoints are increased as a result of the reactor dome pressure increase for the power uprate. The analytical limits for the increased MSRV setpoint pressure are provided in Table 5-1 of Reference 3. The licensee determined that peak RPV steam pressure remains below the ASME allowable of 110% of design pressure and that safety-related MSRV operability is not affected by the proposed changes. The licensee stated that the plant-specific analyses for the 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.

The licensee evaluated MSRV air-clearing loads including MSRV discharge line loads, suppression pool boundary pressure loads and drag loads on submerged structures. An increase in MSRV setpoint pressure due to the power uprate will result in higher opening pressure and, therefore, higher MSRV discharge loads. The increased MSRV loads resulting from the increase in the setpoint pressure were compared with BFN plant unique design limits calculated during the MARK 1 LTP. The licensee indicated that the original MSRV load definition used at BFN is more severe than the conditions for the proposed power uprate. Consequently, the licensee concluded that the SRV loads for the SRV discharge line piping will remain unchanged from the original design basis analysis.

Based on its review, the staff concludes that the MSRVs and the MSRV discharge piping will continue to maintain the structural integrity and to provide sufficient overpressure protection to accommodate the proposed power uprate at Browns Ferry.

In a May 22, 1998, letter, the licensee stated that the system evaluations performed to support the BFN Units 2 and 3 power uprate request considered the effects of increased pressure, flow, and temperatures on the functional performance of safety-related pumps and valves. The proposed power uprate will result in a nominal reactor pressure increase of 30 psi and an increase in main steam flow of 6 percent. The increase in nominal reactor pressure was said to result in a 30 psi increase in the lift setpoints for the MSRVs. Ambient room temperature and maximum post-LOCA containment pressure increases due to the power uprate were determined to be about 3 to 5°F and 1.0 psi, respectively. For conservatism, the licensee included slightly higher increases in the assumed nominal reactor pressure and MSRV setpoint.

The licensee reported that plant systems were evaluated to ensure that pump and valve functional requirements continued to be met with the new power uprate conditions. For example, the licensee evaluated MOVs within the scope of its program developed in response to GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The licensee indicated that four GL 89-10 MOVs were identified for each reactor unit as needing torque

switch adjustments for the power uprate conditions which will be accomplished prior to implementation of the power uprate.

As noted in NRC Inspection Report 50-260, 296/98-03, dated June 16, 1998, the licensee has completed its MOV calculations to support the power uprate for Browns Ferry Unit 3. In a letter dated July 24, 1998, the licensee stated that the calculations for Browns Ferry Unit 2 to reflect the power uprate conditions for safety-related pumps and valves will be completed by August 31, 1998. However, the licensee indicated in its July 24, 1998, letter that calculations to support the required modifications due to the power uprate have been completed. In support of the power uprate, the licensee reported, for Unit 2, that four MOVs will require adjustment of their torque switch. For Unit 3, the licensee stated that two MOVs will require torque switch adjustment and that one MOV will require both torque switch adjustment and spring pack replacement. The licensee also indicated that no adjustments or modifications of other power-operated valves have been found to be necessary to support the power uprate. The licensee will implement these modifications prior to the implementation of the power uprate for BFN Units 2 and 3.

In a July 24, 1998, letter, the licensee stated that it has evaluated its response to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," on pressure locking and thermal binding in light of the planned power uprate. The licensee determined that the previous submittal in response to GL 95-07 remains valid without any necessary revision. The licensee also evaluated the increase in the postulated accident containment temperature for the power uprate condition with respect to its response to GL 89-10, GL 95-07, and GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," regarding potential over-pressurization of isolated piping segments at BFN. The licensee evaluated the effect of the temperature increase on MOV motor output, and the potential increase in valve and piping pressure resulting from the higher containment temperature. The licensee did not identify any adverse impact from the postulated increase in containment temperature. The NRC staff will complete the review of the licensee's response to GL 95-07 and GL 96-06 separate from this power uprate request.

Based on its review of the information provided by the licensee, the staff concludes that the requested power uprate will not have an adverse effect on the performance of mechanical components of safety-related pumps and valves at BFN Units 2 and 3.

(6) CONCLUSION

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

13.0 EMERGENCY OPERATING INSTRUCTIONS (EOIs)

For BFN, the EOPs are designated as EOIs. In the letter dated May 20, 1998, TVA stated that review and revision of the EOIs, which includes a review of all EOI variables and limit curves, for changes due to power uprate, is an ongoing activity during power uprate implementation phase. The review, update of the EOIs for any changes, and training are scheduled to be completed prior to the power uprate start-up for Browns Ferry Unit 3. This is acceptable.

14.0 HUMAN FACTORS ISSUES

The staff's evaluation of human factors issues consist of the following five review topics.

Topic 1 - Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures (EOPs and AOPs). Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will new operator actions be required?

By letter dated April 28, 1998, the licensee stated that its EOPs are symptom-based and that changes to the EOPs and AOPs (Emergency Operating Instructions [EOIs] and Abnormal Operating Instructions [AOIs]) consist of revisions to previously defined numerical values (e.g., reactor pressure vessel high pressure scram setpoint value). The type, scope, and nature of the operator actions required for accident mitigation have not changed with no new operator actions being necessary for power uprate.

The staff finds the licensee's response acceptable.

Topic 2 - Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will affect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (reduced/increased) response times. Discuss why any reduced operator response times are needed. Discuss whether any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to complete the required manual actions in the times allowed. Discuss results of simulator observations regarding operator response times for operator actions that are potentially sensitive to power uprate.

In its letter of April 28, 1998, the licensee identified the following operator actions as particularly sensitive to power uprate: initiating emergency RPV depressurization in response to a loss of high-pressure injection; initiating the SLCS following a high-powered ATWS; and inhibiting the ADS during ATWS. The licensee reviewed its PSA and determined operator response times affecting core damage frequency and compared them to those identified in the GE topical report NEDC-31984P, Supplement 2. The GE report evaluated the impact of power uprate on a generic basis. The report indicated that some operator response times would be reduced after power uprate but the effects on operator reliability and performance were determined to be insignificant. The licensee stated that it reviewed all operator responses used in its PSA and confirmed that the effect on operator response times due to power uprate on Browns Ferry was consistent with the GE generic findings.

For the three actions sensitive to power uprate, the licensee compared pre-and post-power uprate times (assumed from the PSA). The licensee indicated that, for the action of recognizing the need to depressurize, which had a preuprate response time of 30 minutes, there would be a postuprate reduction of 2 minutes in operator response time and a 6-second reduction in response time from the 3-5 minute preuprate response time to reach the (-)190 inch vessel level (once the (-)162 level was reached). For the operator action of initiating SLCS given an ATWS with the vessel isolated, under preuprated conditions, the operator had 2 minutes to reach the boron injection initiation temperature (BIIT); under uprated conditions, the operator's response time would be reduced by 6 seconds. For the operator action of inhibiting ADS during ATWS, the preuprate operator response time was 3.5 minutes; the uprate operator response time was reduced by 6 seconds. The licensee stated that, because these operator actions are controlled by EOPs, the slight reduction in response times noted for the power uprate condition "will not significantly affect the operator's ability to safely complete the required actions."

While the staff found that the minimal reduction in response times should not significantly affect the operator's ability to complete the actions, the staff had concern that those actions assumed to be performed in 5 minutes or less by the PSA may be unachievable under realistic conditions. ANSI/ANS Standard 58.8, "Time Response Design Criteria for Safety-Related Operator Actions," 1984), indicates that for events that occur with an estimated frequency of @10^{-1} per reactor year (e.g., reactor coolant system depressurization), a minimum of 5 minutes is allowed for the operator to diagnose the event (i.e., verify automatic responses, observe plant parameters, plan subsequent actions) or the safety-related systems and components "shall be initiated by automatic protection systems" (less frequently occurring events require more diagnosis time). Therefore, the staff requested that the licensee provide supporting information that operators can perform the required tasks under simulated accident conditions in the times assumed by the PSA.

In its July 24, 1998, response, the licensee clarified that "the actions specified to occur within 5 minutes are all in response to an ATWS event." EOI's, the licensee's emergency operating procedures, are used by the operator to complete the steps, "with no other required actions [to] distract the operator from completing the necessary mitigating actions and the accomplishment of the assigned duties." They further state that, for initiating SLCS pump start and inhibiting the ADS, two of the actions of concern, " ... are performed from direct procedural guidance and both can be completed using switches immediately available to reactor operators in the main control room's control area. SLCS pump start is accomplished by turning the SLCS keylock switch (key located next to switch on panel) to either 'Start A' or 'Start B' position. Inhibiting ADS is accomplished by turning two keylock switches (keys located next to switches on panel) to 'Inhibit' position. These are straightforward operator actions and, thus can be performed rapidly." Manually opening individual MSRVs, a third action of concern, is accomplished from the main control room by "placing the control switch to the 'Open' position. This is a straightforward action and, thus, can be performed rapidly." The licensee also clarified that the values for operator actions assumed by the PSA were based on:

> Detailed human action evaluation work [was] accomplished to assign an unique numerical value for selected human actions including the actions associated with dynamic actions accomplished during the plant response to an initiator. These human actions evaluations were accomplished by including an interview with operating personnel [and] guided by selected factors [including]: a) plant man-machine interfaces and indications of conditions;

b) task complexity; c) adequacy of time to accomplish the actions; d) significant preceding and concurrent actions; e) stress; f) training and experience, and ; g) procedural guidance. The interviewees were encouraged to utilize simulator experience and observations as input to the question responses.

The staff finds the licensee's response to the staff concerns acceptable.

The licensee also indicated that one manual action (termination of the HPCI system injection following spurious initiation during an Appendix R fire event covered by their Appendix R Safe Shutdown Instructions (SSIs)) has a reduction in response time of 3 minutes (i.e., from a predicted time of 10 minutes to 7 minutes). This operator action is necessary to prevent flooding of the main steam lines. The licensee explained that the reduction in response time is not a result of the power uprate but rather a result of using different models to predict pre- and post-uprate operator action times (i.e., the GE SAFE model for preuprate predictions and the SAFER model for post-uprate).

The licensee further indicated that this operator action of shutting down HPCI is required in nine Unit 2 SSIs and seven Unit 3 SSIs. One instruction for each Unit requires HPCI shutdown from outside the main control room. The licensee stated that this action (close one valve from the 250V DC reactor motor-operated valve board located just outside the main control room on the same elevation), has been performed in a shorter time than allowed by the SAFER model (i.e., seven minutes).

The staff requested the licensee to explain the statement, on page E1-4 of its letter dated April 28, 1998, "TVA has previously demonstrated that this action, close one valve from the 250V DC reactor motor operated board, located on the same elevation just outside the main control room, can be performed within the shorter time predicted by the SAFER model." The staff asked, when was this demonstration accomplished and how was it accomplished. Were the following factors considered: environmental conditions expected; procedural guidance for the required actions; support personnel and/or equipment required to carry out the required actions; information requirements including qualified instrumentation.

In response to the staff's questions related to manually isolating HPCI, the licensee stated in its July 24, 1998, letter that,

> The Appendix R fire event which requires manual isolation of the HPCI from outside of the control room is a fire in fire zone 16, which is a control building fire in plant elevation 593 though plant elevation 617. ... TVA Operations personnel performed in plant drills on October 25, 1995 to validate all manual operator action times. Drill performance was observed by the NRC and documented by NRC as part of an NRC inspection (NRC Inspection Report 95-60).

The licensee continues by stating that, "For Unit 2, the 10 minute HPCI isolation steps were 33 and 34 of 36 total action steps for the associated Unit 2 procedure. The entire associated procedures performance was completed in 8 minutes, 20 seconds." All actions of the procedure were completed at the control boards in the same Shutdown Board Room, except for travel from the main control room and thus, the licensee indicates that each step of the procedure takes approximately the same amount of time. For the power uprate, the licensee

indicates in its July 24, 1998, letter, that the procedure is being revised, moving the steps which isolate HPCI to the first steps of the procedure. The licensee states that, as a result of the procedure reconfiguration, "this action will be performed in considerably less than 7 minutes."

For Unit 3, the 10 minute isolation steps were steps 10 and 11 of 62 total steps of the associated procedure. Performance of the entire procedure took 12 minutes and 15 seconds. However, all actions of this procedure are not completed in the same Shutdown Board Room, with travel from the Shutdown Board Room to locations ending in the reactor building adding "considerable time not used in performing control switch manipulations for this procedure." However, the licensee states in its July 24, 1998, letter, that the actions to isolate HPCI for Unit 3 "will be performed in considerably less than 7 minutes" and, as with the Unit 2 procedure, the Unit 3 procedure is being revised to move the steps isolating HPCI to the first steps of the procedure. They further describe that the Shutdown Board Room, from which the manual action to isolate HPCI is performed, is protected by a safety-related vent system and is independent from the fire zone. No new controls, special equipment or additional support personnel are required to take the required actions.

The staff finds the licensee's response to the staff concerns acceptable.

The staff also asked for clarification related to the SAFER model referred to by the licensee in its April 28, 1998, response to the staff's RAI, used to analyze LOCA conditions and fuel heatup activities. In its "Power Uprate Safety Analysis for the Browns Ferry Nuclear Plant Units 2 & 3" report (enclosure 5 of the licensee's October 1, 1997, submittal), GE applied the model to analyze an Appendix R fire event and stated that, "Sufficient time is available for the operator to perform the necessary actions" (p.6-9). The staff asked how the SAFER model evaluates human actions and how did GE conclude that operators have sufficient time to perform necessary actions.

The licensee, in its July 24, 1998, letter, clarified that the SAFER model does not evaluate human actions and that the demonstration of operator response times is covered in responses to previous staff questions.

The staff finds the licensee's response to the staff concerns acceptable.

The staff further asked for clarification on the SAFER model being used for the power uprate analysis and the SAFE model for the preuprate analysis. The SAFE model, according to the licensee in its April 28, 1998 submittal, predicted a required operator action time to shut down HPCI of 10 minutes; the SAFER model predicted 7 minutes to shut down HPCI. Which action time is TVA taking credit for?

The licensee stated in its July 24, 1998, letter, that it will implement the 7 minute action time.

The staff finds the licensee's response to the staff concerns acceptable.

Lastly, the staff asked the licensee to address the question of what are credible errors that operators could make in shutting down HPCI (the action as required from the control room and from outside the main control room). What are the consequences of the operator failing to accomplish the action and how will recovery from the failure(s) be accomplished? How does

TVA know that operators can successfully recover from credible errors, i.e., provide evidence that operators can recover from credible errors?

In its July 24, 1998, letter, the licensee explained that "no changes are being made in the design approach of the Appendix R program due to the power uprate. The procedural step changes being made do not introduce any new human factors engineering issues." In addition, operator requalification training with detailed procedures, control board layout and location, "provide a combined influence for a low probability of operator error." The licensee maintains that, in the event of an operator error in the performance of this step, "or any other error affecting reactor control parameters, reactor water level, or reactor pressure, resulting conditions would be recognized by the operators using indications at the backup control panel."

The license indicated that consequences of not making the 7 minute HPCI time action could potentially fill the reactor vessel up to the main steam lines and result in RCIC being disabled by liquid carryover. If the action were not performed within the specified time limits, "recovery actions by the operators would be to drain RCIC and utilize other alternate injection paths in that time frame. High confidence exists that this is a very recoverable scenario... ."

The staff finds the licensee's response to the staff concerns acceptable.

Topic 3 - Discuss all changes the power uprate will have on control room alarms, controls, and displays. For example, will zone markings on meters change (e.g., normal range, marginal range, and out-of-tolerance range)? If changes will occur, discuss how they will be addressed.

In the licensee's April 28, 1998 submittal, it stated that two changes in control room indications will result from the planned power uprate. Three control room indicators (wide range reactor pressure) will be modified to revise the low limit of the indicator scale's "red zone" to accommodate the increase in reactor pressure scram setpoint as a result of the power uprate. Also, forebay temperature indication span will be modified to accommodate a lower forebay high temperature setpoint.

The licensee also indicated that various main control room instructional aids will be revised because of the power uprate. These aids include labels, sketches, or markings which are posted and used as memory aids or instructional guidance. Power/flow map, reactor water level compensation curves at reduced pressure conditions, and main steam relief valve setpoints/locations will also be modified to address the power uprate. All changes will be prepared in accordance with plant modification procedures which includes detailed review of the proposed control room design change package and all required changes will be implemented before operating the plant at uprated power.

The staff finds the licensee's response acceptable.

Topic 4 - Discuss all changes the power uprate will have on the Safety Parameter Display System (SPDS) and how they will be addressed.

The licensee indicated in its April 28, 1998, submittal that the design and intent of the SPDS will remain unchanged. The information presented on the SPDS top-level display and the method of presentation remain unchanged. Changes to SPDS include recalibration of input/output

points; changes to constants which input to the displayed points (e.g., rated core thermal power); changes to MSIV lift setpoints; and changes to EOI limit graphs. The licensee stated that these changes will be transparent to the operator because EOI execution is unaffected.

The staff finds the licensee's response acceptable.

Topic 5 - Describe all changes the 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.

The licensee indicated the following in its April 28, 1998, submittal:

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

The licensee stated that operator training content will be revised to include the power uprate and that classroom and simulator training will be conducted prior to implementation of the power uprate on Unit 3. Additional training will be provided as deemed necessary by the Power Uprate Refueling Test Program. The licensee will compare non-uprate to uprate data sets for major simulator transients.

(b) Complete simulator changes that are consistent with ANSI/ANS 3.5-1985.

The licensee stated that simulator fidelity will be re-validated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." Simulator re-validation 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 that simulator changes will be implemented prior to installation of power uprate modifications in the plant. Two software loads will be maintained, one for power uprate changes and the other for Unit 2 configuration and non-uprate conditions. Differences will be recorded in a Unit 2/Unit 3 Differences listing, which will be incorporated in the training program. Simulator re-validation in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, will be accomplished in two stages. First, the simulator performance will be validated against the power uprate design and engineering analysis data. Second, once plant modifications are completed on Unit 3, data will be collected and compared with simulator performance data. TVA intends to integrate this re-validation with the BFN simulator quadrennial testing and certification program.

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

See evaluation of Topics 3 and 4, above.

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

The licensee stated that the simulator changes will be integrated with the BFN simulator quadrennial testing and certification program. This program is understood to provide administrative controls, practices, and procedures sufficient to reflect issues and discrepancies identified during startup.

In its April 28, 1998, letter, the licensee made the following commitments with respect to simulator modifications, and classroom and simulator training:

1. Operator training content will be revised to include power uprate. Classroom and simulator training will be conducted prior to restart of Unit 3 at uprated conditions.

2. The plant simulator control room and plant simulator process computer changes will be completed as required to support the training program. TVA expects to complete this action prior to simulator training for power uprate. By letter dated September 1, 1998, TVA stated that this action has been completed.

3. An acceptance test will be run to benchmark simulator performance based on design and engineering analysis data. The primary focus will be matching the heat balance data for the 105 percent power steady-state test as required in ANSI/ANS 3.5-1985, Appendix B1.1. TVA expects to complete this action prior to simulator training for power uprate.

4. Once the power uprate modifications are completed on Unit 3, data will be collected, including the post power uprate refueling test report, and compared to simulator data as required by ANSI/ANS 3.5-1985, Section 5.4.1. TVA expects to complete this as part of BFN's required simulator certification program. TVA expects to submit this report within 60 days following completion of certification.

5. Operator training will be modified, as necessary, based on observations of plant and operator performance during the Power Uprate Refueling Testing Program.

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 power uprate. The staff finds the licensee's response acceptable. These commitments, as identified in Items (1) through (5), above, have been incorporated into the Facility Operating Licenses for both units as additional License Conditions.

Based on the above discussion, the staff concludes that (a) the analyses and evaluations supporting the power uprate follow the generic guidelines approved by the staff and hence is acceptable and (b) the power uprate for the BFN units 2 and 3 should not adversely affect operator actions or operator reliability.

(15) <u>TEST CONTROL</u>

(1) Regulatory Basis

Regulatory provisions for the testing of structures, systems, and components are identified under Criterion XI, "Test Control," of Appendix B to 10 CFR 50. The program for implementing these requirements is described in section 9.4 of the TVA Nuclear Quality Assurance Plan

(NQAP), Reference 6. The NQAP description follows the guidance of ANSI N45.2-1971, section 12 (Reference 7) with respect to the development of test procedures, conduct of testing, and documentation and evaluation of test results. Test results are documented in a suitable test package, including deviations and adverse conditions, and actions taken to resolve the condition. Documentation of test results are maintained as lifelong records, in accordance with the guidelines provided by Regulatory Guide 1.28, Revision 3 (Reference 8), to which the licensee has committed.

(2) Generic Test Guidelines for GE BWR Power Uprate

NEDO-31984, section 5.11.9 provides the general guidelines for power uprate testing.

> A testing plan will be included in the uprate licensing application. It will include pre-operational tests for systems or components which have revised performance requirements. It will also contain a power increase test plan.

Guidelines to be applied during the approach to and demonstration of uprated operating conditions are provided in section L.2 of GE proprietary report NEDC-31897P-1. GE report NEDC-32751P, submitted with the licensee's application, provides the required additional information relative to power uprate testing.

(3) Startup Test Plan

The license will conduct limited startup testing at the time of implementation of power uprate. The tests will be conducted in accordance with the guidelines of NEDC-31897P-1 to demonstrate the capability of plant systems to perform their designed functions under uprated conditions.

The tests will be similar to some of the original startup tests, described in section 13.5 of the licensee's UFSAR. Testing will be conducted with established controls and procedures, which have been revised to reflect the uprated conditions. Revised plant procedures, reflecting the uprate conditions, will be used to the extent practicable during the test program.

The tests consist essentially of steady state, baseline testing between 90 and 100 percent of the currently licensed power level. At least one set of data will be obtained between 100 and 103 percent power, and a final set of data at the uprated (105%) power level. The tests will be conducted in accordance with existing site procedure SSP 8.1, "Conduct of Testing," which implements the licensee's QA program test control requirements.

The following power increase test plan is provided in NEDC-32751P, section 10.5, "Required Testing."

(a) Surveillance testing will be performed on the instrumentation that requires recalibration for power update.

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

(c) Power increases beyond the previous rating will be made in increments along an established flow control/rod line. Steady state operating data, including fuel thermal margin, will be taken and evaluated at each step.

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

The licensee's test plan follows the guidelines of NEDC-31897P-1 and the staff position regarding individual power uprate amendment requests.

(4) Performance Testing

(a) Systems/Components with Revised Performance Requirements

NEDO-31897, section 5.11.9 guidelines specify that pre-operational tests will be performed for systems or components which have revised performance requirements. The licensee's submittal, NEDO-32751P, identifies the following systems and components that have revised performance requirements.

- Reactor Recirculation System (RRS) - The RRS system operating pressure, temperature, drive motor horsepower, pump flow, and pump brake horsepower will be increased.

  These tests will demonstrate flow control over the entire pump speed range to enable a complete calibration of the flow control instrumentation. The tests will also demonstrate acceptable reactor internal vibration levels at the uprated conditions.

- RCIC - The RCIC pump maximum specified pump and turbine rated speed will increase in order to deliver its design flow rate at the higher pump discharge pressure.

- HPCI - The HPCI pump maximum specified pump and turbine rated speed will increase in order to deliver its design flow at the higher pump discharge pressure.

  RCIC and HPCI tests will demonstrate that they are capable of meeting flow requirements at the increase pump discharge pressures. These tests will also demonstrate that the reliability of these systems are not decreased by the higher loads placed on the systems or because of any system modifications to compensate for the increased loads. Since theses systems are within the scope of the Maintenance Rule, system reliabilities will be established and tracked.

- Instrument setpoints (e.g., high pressure scram) will be established based on the power uprate safety analyses. New analytic limits for setpoints, including reactor high pressure trip and ATWS high pressure recirculation pump trip will be established.

- Pressure Control System (PCS) - The PCS consists of the pressure regulation system, turbine control valve system, and steam bypass valve system. Power uprate testing will confirm that the pressure regulating tuning parameters set prior to startup are acceptable, or test results will indicate what tuning parameter adjustments are necessary.

- Turbine Generator - The increased reactor operating pressure associated with uprated conditions has the potential to result in increased turbine overspeeding during system startup, increasing the probability of the system to trip during startup.

- SLCS - The pressure at which the system injects will be increased.

- MSRV - The MSRV set point pressure will be increased.

- EDGs - Some additional loads may be added.

- MOVs - Pursuant to the guidance of GL 89-10, modifications may be necessary to some motor operated-valves.

(b) Planned Performance Tests

The licensee plans to conduct tests during the ascension to power uprate conditions. The performance tests and associated acceptance criteria are based on the Browns Ferry original startup test specifications and previous GE BWR power uprate test programs. The licensee has identified performance tests for the following systems:

- Reactor Recirculation
- RCIC
- HPCI
- Electro-Hydraulic Pressure Control
- Turbine Generator
- SLCS
- Reactor Water Level Measurement
- Feedwater Control
- Drywell Atmosphere Cooling
- Intermediate Range Neutron Monitoring
- Average Power Range Monitoring

With regard to systems and components with revised performance requirements identified in the preceding section, but not identified above:

- MSRVs will be reset at Wyle Laboratories in accordance with the inservice inspection program (see revised surveillance requirement SR 3.4.3.1).

- EDG - The power uprate analysis, performed subsequent to the licensee's submittal, concluded that sufficient margin existed in diesel generator capacity that the additional electrical loading would have an insignificant impact on the station load flows. (See licensee's letter dated May 20, 1998, Section B.2). EDG testing conducted under SR 3.8.1.7, provides assurance that the diesels can fulfill their design requirements; no special testing is planned as part of the startup test program.

- MOV testing - Post-modification testing of motor-operated valves modified as a result of the power upgrade will be conducted as part of the design change control closure process; no special testing is planned as part of the startup test program.

The planned performed tests follow the guidelines of NEDC-31897P-1 with respect to performance testing of systems or components that have revised performance requirements.

(5) Conclusions

The licensee's program for startup testing follows the guidelines of NEDC-31897P-1, which has been accepted by the NRC as the generic basis for power uprate amendment requests. The submittal provides a test program that follows NEDC-31897P-1 guidelines for uprate testing and meets 10 CFR 50, Appendix B requirements for test control. The licensee's power uprate test program is acceptable.

16.0 EVALUATION OF CHANGES TO FACILITY OPERATING LICENSE AND TS

As a result of the power uprate, the licensee proposed following TS changes:

(1) TS 1.1, Definitions

Revise the RTP value from 3293 to 3458 MWt to reflect increased power level.

(2) SR 3.1.7.6, and TSB 3.1-44

Revise the value of SLCS pump discharge pressure from 1275 to 1325 psig to account for the effects of increased main steam relief valve setpoints.

(3) Table 3.3.1.1- 1, Function 2.b

Revise the allowable value for APRM Flow Biased Simulated Thermal Power - High Scram setpoint intercept from 71 percent to 66 percent RTP to maintain the preuprate power/flow relationship in terms of absolute power.

(4) Table 3.3.1.1- 1, Function 3

Revise the allowable value for Reactor Vessel Steam Dome Pressure - High scram setpoint from 1055 to 1090 psig to account for increased 30 psi reactor operating pressure.

(5) SR 3.3.4.2.3b

Revise the allowable value for Reactor Steam Dome Pressure - High Anticipated Transient Without Scram-Recirculation Pump Trip (ATWS RPT) setpoint from 1146.5 to 1175.0 psig to account for increased 30 psi reactor operating pressure.

(6) (For Unit 2 Only) Table 3.3.6.1-1, Function 5

Revise the allowable value for RWCU system isolation to 188°F

(7) Figure 3.4.1-1, TSB 3.4-3, Applicable Safety Analyses

Revise lower bounds (rod lines) for Thermal Power vs Core Flow Stability Regions from 108%, 100% and 80% to 102.9%, 95.2% and 76.2% respectively, to maintain the preuprate power/flow ratios in terms of absolute power.

(8) SR 3. 4. 3. 1

Revise the MSRV setpoints from 1105, 1115 and 1125 to 1135, 1145 and 1155 respectively, to account for increased 30 psi reactor operating pressure.

(9) LCO 3.4.10, SR 3.4.10.1

Revise the LCO and SR for reactor steam dome pressure to be less than 1050 psig.

(10) SR 3.5.1.7, TSB 3.5-12, SR 3.5.1.7, SR 3.5.3.3, TSB 3.5-28, SR 3.5.3.3

Revise upper and lower bounds of HPCI, and RCIC pump test pressure from 1010 and 920 psig to 1040 and 950 psig, respectively, to account for increased 30 psi reactor operating pressure.

(11) LCO 3.7. 1, Condition D, SR 3.7.1.2 p. 3.7-3a, Figure 3.7.1-1 p. 3.7-5, SR 3.7.2.1 TSB 3.7-1, Background, TSB 3.7-2, Applicable Safety Analyses, LCO, Applicability p. B 3.7-5, Actions, TSB 3.7-8, LCO

Add LCO and Action statement addressing UHS for RHRSW. Add upper limit for the UHS average water temperature for RHRSW to be in accordance with new Figure 3.7.1-1. The value is used for calculating long-term suppression pool temperature response in the new containment analysis. Add a note to for the UHS average water temperature for ECCW to refer to additional UHS requirements from RHRSW.

TSB 3. 5-4, Background, TSB 3.5-24, Background

Revise value of upper design pressure for HPCI and RCIC operating range 54 psi from 1120 to 1174 psig to account for the effects of increased main steam relief valve setpoints.

TSB 3. 6-2, TSB B 3. 6-7, Applicable Safety Analyses

Revise value of P., (from 49.6 to 50.6 psig) to account for results of the new containment analysis.

TSB 3.7-2 B 3.7.1 B 3.7-6, References

Revise flow rate for RHRSW (from 4.500 to 4000 GPM) used in peak containment temperature analysis. Revise peak containment pressure (from 49.6 to 50.6 psig) to reflect new analysis results. Revise reference for peak suppression pool temperature.

As discussed in Section 2.0 of this SE, TVA's planned approach for achieving the higher power 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 the feedwater flow, (3) no increase in the maximum core flow, (4) a small (less than 3 percent) increase in the reactor operating pressure, and (5) reactor operation primarily along extensions of pre-uprated rod/flow control lines. The operating pressure will be increased approximately 30 psi to assure satisfactory pressure control and pressure drop characteristics for the increased steam flow. For the reasons discussed in Sections 2 through 16 of this SE, the staff finds the above system design and performance conditions will not adversely impact the safe operation of the plant at the increased power level, and therefore, acceptable. The staff also finds reasonable assurance that the radiological consequences of normal operations and anticipated accidents at the BFN Units 2 and 3 will continue to meet regulatory requirements at a rated thermal power of 3458 megawatts and, therefore, are acceptable. The staff has reviewed the licensee's proposed TS changes and determined that they reflect operation at the increased power level, and therefore, acceptable.

17.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official (Kirk Whatley) was notified of the proposed issuance of the amendment. The State official had no comments.

18.0 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 September 1, 1998, (63FR46491).

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

19.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

20.0 REFERENCES

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

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

3. GE Nuclear Energy, "Power Uprate Safety Analysis for the Browns Ferry Nuclear Plant Units 2 & 3," Licensing Topical Report NEDC32751P, Class III (Proprietary), September 1997.

4. GE Nuclear Energy, "Browns Ferry Nuclear Plants Units 1, 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (Revision 1)," NEDC-32484P, February 1996.

5. SECY-91-401, "Generic Boiling Water Reactor Power Uprate Program," December 12, 1991.

6. TVA Quality Assurance Plan (TVA-NQA-PLN89-A), Revision 7, submitted by letter dated October 11, 1996.

7. ANSI N45.2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants," October 1971.

8. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction," Revision 3, 1985.

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the proposed uprated power level will have no impact on the EQ of safety-related mechanical components inside or outside the containment and, therefore, is acceptable.

11.0 ELECTRICAL ISSUES

The staff has reviewed the main generator and its auxiliary equipment, electrical power systems, and safety-related electrical equipment environmental qualification to determine if the power uprate would have any adverse impact on the existing (onsite and offsite) power system and to see if any TS changes are necessary.

(1) Main Turbine Generator and its Auxiliary Equipment.

The licensee has evaluated the power uprate of the main generator and its auxiliary equipment. The licensee reviewed the following components:

- Main Turbine Generator: For the proposed uprate of the original core thermal power, the turbine throttle pressure is required to increase from 980 psia to 1010 psia. Since the change in heat load from the current generator operating condition to the uprate condition is already factored into the design of the generator and its auxiliary equipment, and since the generators will be operating within the originally designed capability curves, no hardware changes are necessary. Thus, the licensee finds that the main generator has the capability to support the power uprate.

- Generator (Stator and Rotor) Cooling Systems: With power uprate, the generator stator bar temperature would increase without any additional cooling water flow. The licensee has evaluated the performance of the generator stator water cooler and the adequacy of its in-service stator winding assembly. The licensee has also reviewed the existing generator's rotor hydrogen coolers. Since the original generator is rated 1280 MVA at a power factor of 0.93 (i.e., 1190 MW), the proposed electrical output of 1156 MW under power uprate is still less than the originally rated generation output of 1190 MW. Therefore, the licensee finds that the generator's cooling systems are capable of reliable operation for the uprated heat load and expects no adverse impact on heat loads as a result of the power uprate condition.

- Exciter Cooling System: The licensee has reviewed the generator excitation system that consists of the exciter, voltage regulation, and rectifier, for the power uprate conditions and determined the performance of the exciter cooling system at the power uprate conditions to be adequate to handle the associated heat load.

- High-Voltage Bushings, Current Transformers, and Generator Protective Relays: For power uprate, the licensee has reviewed the in-service, high-voltage bushing and current transformers and determined that an adequate design margin exists to handle the associated heat load and increased electrical capacity. The generator protective relays have been evaluated as part of the grid stability at the new power uprate condition. The results indicate that the existing generator protective relays' performance is acceptable for power uprate.

Because the original design of the main generator has adequate capacity to handle the additional heat loads, produced by the power uprate conditions of the main generator, stator, rotor, excitation systems, bushings, and current transformers, the licensee concludes that no modifications are needed to the main generator and its auxiliary equipment. The staff finds this acceptable.

(2) Electrical Power Systems

Although no electrical hardware needs to be replaced for power uprate, it is necessary to assess the increase in the electrical power requirements for the condensate, condensate booster, and recirculation pumps in order to demonstrate that the increased load of these electrical components for power uprate has a minimal effect on BFN's electrical onsite power systems and that the increased load is well within the existing analyzed capability of the plant's onsite electrical distribution system. The licensee identified the following: (1) the condensate pump load would increase from 849 hp (preuprate) to 867 hp (power uprate), (2) the condensate booster pump load would increase from 1519 hp to 1559 hp, and (3) the recirculation pump load would increase from 8378 hp to 8678 hp. However, the licensee states that the electrical calculations for the auxiliary power system were performed using nameplate data for those affected loads (i.e., 900 hp, 1750 hp, and 9000 hp, respectively). Since the nameplate loads are higher than the operating loads at either preuprate or uprate conditions, the licensee concluded that the calculations for the preuprate are the same as for the uprate; thus, there is no need for further analysis for the proposed power uprate. On this basis, the licensee finds that the current 4-kV and the 480-V onsite power distribution system should remain adequate for the power uprate conditions.

With an increase of 57.5 MW generation to each unit, the licensee has evaluated the impact on the following components:

- Main Power Transformer (MPT): The MPT was evaluated against the maximum rated generator power production expected for the power uprate condition to establish the minimum generator power factor that could be tolerated without exceeding the transformer's 1200 MVA rating. For the proposed electrical power uprate, the licensee has determined that a generator power factor of .94 and .95 would be required for Units 2 and 3, respectively. Considering the plant house loads (through the unit station service transformers) for the onsite distribution system during normal plant operation, the MPT loadings during power uprate conditions for both units would not exceed the 1200 MVA rating of the MPT.

- Emergency Diesel Generators (EDGs): As part of the power uprate implementation, the licensee has reviewed whether all GL 89-10 MOVs would be impacted by power uprate. The results of the GL 89-10 study showed that no motor changes are required for MOVs, but the replacement of one torque switch and the reset of three other torque switches will be needed for power uprate implementation. With such minor changes for MOVs, the licensee concluded that no plant electrical equipment changes are necessary to support power uprate. Therefore, the licensee finds that there would be no impact on the EDG loading, as it would remain the same for power uprate, thus sufficient capacity exists for EDGs.

- Grid Stability: The licensee has studied generator stability associated with the increased generator capacity and has established minimum gross MVAR limits for each unit. For certain BFN 500-kV line outages, the above MVAR limitations would be incorporated into the BFN Operations Standing Orders as part of the implementation of power uprate.

- Grid Voltage and Degraded Grid Voltage Relay Setpoints: With additional generation to the BFN offsite system, the licensee has evaluated how this additional generation would affect the grid voltages and assessed whether new grid voltage would change the current grid system voltage profile, so that it affects the degraded grid voltage (DGV) relay setpoints. With preuprate switchyard voltage calculated to be 525.9 kV (nominal 500 kV) with both BFN units operating at 1100 MW, the switchyard voltage for power uprate is calculated to be 524.1 kV with both units operating at 1180 MW, which exceeds the proposed power increase of 57.5 MW. The grid system voltage profile change from preuprate to uprate conditions is 0.34%, and it has negligible impact to the plant's auxiliary power system. The licensee finds that since the grid voltage change is minimal and the methodology and software to perform the PSB-1 analysis have not changed over the years, it is not necessary to determine a new DGV relay setpoint because the DGV relay setpoints would remain the same for the power uprate.

- Isophase Bus: The isophase bus systems for BFN Units 2 and 3 are currently (preuprate conditions) carrying 32,376 amps. With the proposed uprated condition, the systems would be expected to carry 33,591 amps. Since the isophase bus cooling systems are rated for 35,270 amps, the licensee finds no problems for the isophase bus to carry the uprated 33,591 amps.

In addition, as a result of lessons learned from the Maine Yankee Independent Safety Assessment Inspection, all licensees are required to review and evaluate whether the power uprate would alter the original licensing basis for GDC-17 and the SBO requirement.

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

Plant response and coping capabilities for an SBO event could be affected by operation at the uprated power level because of the increase in the operating temperature of the primary coolant system, increase in the decay heat, and increase in the main steam safety relief valve setpoints. The licensee has re-evaluated the SBO requirement using the guidelines of NUMARC 87-00. The licensee found that: (1) the temperature increases in the control and relay rooms are not affected by power uprate; (2) the HPCI and RCIC equipment room temperature responses increase, but are within existing margins; and (3) the drywell area temperatures increase by a little amount, but the equipment necessary for event mitigation is qualified for these temperatures. Although the water requirement for the condensate increases, the current design of the condensate storage tank ensures that adequate water

volume is available. This ensures the RCIC operation throughout the coping period. On this basis, the licensee concluded that the plant will continue to meet the requirements of SBO after power uprate.

Based on the review of the licensee's evaluation, the staff finds that the electrical power system will continue to perform its intended safety-related functions for the 5% power uprate. The staff concludes that the impact of the load increase to the station auxiliary electrical distribution system is not adversely impacted by the proposed power uprate at BFN.

(3) Safety-Related Electrical Equipment Qualification

For power uprate, the licensee has evaluated the Equipment Qualification Data Packages (EQDPs) that document the qualification of safety-related electrical equipment currently installed at BFN for normal, abnormal, and accident environments. The licensee has reviewed the following areas for environmental changes on the inside and outside of the primary containment: (1) pressure, (2) temperature, (3) radiation, (4) humidity, and (5) submergence.

For inside the primary containment, short-term and long-term (100 day) containment response analyses were performed for both the design-basis LOCA and the bounding MSLB by using the GE proprietary computer codes M3CPT and SHEX. For outside primary containment, the HELB analyses were performed using the GOTHIC computer code for power uprate, and the preuprate analyses were performed using the MONSTER computer code.

The radiological analyses for power uprate are based on source-term inventories generated using the ORIGEN methodology. The preuprate radiological doses were calculated using source-term inventories based on TID-14844.

(a) Inside the Primary Containment

- Pressure: The most limiting event for drywell pressure is the design-basis LOCA for which the peak containment pressure increased from 49.6 psig to 50.6 psig with power uprate. Since the safety-related electrical equipment within scope of the BFN 10 CFR 50.49 EQ program has been evaluated for a peak containment pressure of 55 psig, the licensee found no impact on the pressure evaluations for equipment located inside the primary containment.

- Temperature: Since the normal operating temperature for the reactor vessel will increase by less than 4°F for power uprate, the licensee states that changes to normal and abnormal temperature inside the containment are expected to be small. The licensee has evaluated the normal and abnormal temperatures currently used for qualification of electrical equipment located inside the containment and determined that they bound the temperatures expected during operation at power uprate conditions.

  For the accident environment, the peak drywell temperature for a design-basis LOCA increased slightly from 295°F to 297°F; the limiting event for drywell temperature is the MSLB, for which the peak temperature increased from 322°F to 336°F. With the increased peak post-accident temperature in the drywell, the margins between the peak accident temperatures and the peak test temperatures were reduced and, in some cases, the

margin for the peak temperature was shown to be less than the 15°F suggested by IEEE Standard 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." However, although the peak containment temperature increased for the MSLB, the duration of the MSLB temperature transient is shorter and the long-term temperatures are lower than in the preuprate MSLB profile. Therefore, the power uprate MSLB profile inside the containment is actually less severe for post-accident operability than the profile used for preuprate EQ evaluations. For those cases in which the equipment test profile does not envelop the plant accident profile for the entire post-accident time, the licensee performed calculations using the Arrhenius methodology that show an increase in margin when compared to the preuprate calculations for the same equipment.

- Radiation: There is no change to accident radiation doses inside the primary containment as a result of the change from TID-14844 to the ORIGEN methodology. Although the power uprate radiological evaluations indicate that the normal radiation would increase inside the primary containment, the demonstrated (by test) dose for the safety-related electrical equipment located inside the containment was determined to be greater than the total integrated dose (normal plus accident) including margin. Therefore, the licensee found that there is sufficient margin available to accommodate the increased uprate dose.

(b) Outside the Primary Containment

- Pressure: The GOTHIC analysis results indicate that the peak pressures for HELBs outside of the primary containment are lower than the peak pressures from the preuprate MONSTER analysis. Therefore, the licensee found that there is no impact on the pressure evaluation previously performed for electrical equipment located outside of the primary containment.

- Temperature: The peak reactor building temperatures for HELB events calculated using the GOTHIC computer code were higher for certain areas than the preuprate peak temperatures calculated using the MONSTER computer code. However, the margins between the peak accident temperatures and the peak test temperatures for the safety-related electrical equipment were greater than the 15°F suggested by IEEE Standard 323-1974 with the exception of certain Limitorque motor operators in the torus room. The technical justification for accepting the less than suggested 15°F margin is based on a review of the EQ test report. The report indicates that the motor operators were tested at 250°F for more than 24 hours. Although the plant profile reaches a peak temperature of 250°F, the duration of the accident temperature transient is extremely short and the room temperature decreases to less than 180°F in 1 minute. The power uprate HELB profiles outside the primary containment are generally less severe for post-accident operability than the profiles used for preuprate EQ evaluations. Calculations performed using the Arrhenius methodology show an increase in margin when compared to the preuprate calculations for most equipment.

- Radiation: The normal operating doses for most reactor building areas were bounded by the normal radiation doses used for previous equipment qualification. In a few areas, the normal radiation doses increased as a result of operation at power uprate conditions. The only area in which the accident radiation dose increased above the dose used for previous

EQ evaluations was in the standby gas treatment building. The licensee has evaluated these radiation dose increases and found that sufficient margin exists since the total integrated dose, including margin, remains less than the demonstrated (by test) dose for the affected equipment.

- Humidity: The licensee found that there is no change to the normal, abnormal, or accident humidity values for power uprate conditions for inside and outside the primary containments.

- Submergence: The licensee found that there is no significant change to flood levels for power uprate conditions for inside and outside the primary containments.

Since the licensee's evaluation for EQDPs has not identified conditions at the uprate power level that would exceed the conditions to which the equipment was tested, the staff concludes that the safety-related electrical equipment currently installed at BFN is qualified for normal, abnormal, and accident environments at the power uprate conditions.

However, the staff noted that a degraded equivalency analysis using the Arrhenius methodology was performed to extend the test duration to the BFN-specific accident requirements. As a separate initiative outside the scope of this evaluation, the staff will continue to review the adequacy of degraded equivalency analysis using the Arrhenius methodology.

(4) Conclusion

The staff concludes that the proposed power uprate at BFN would have no adverse impact on the station's electrical system and does not require any TS change for the electrical power system.

12.0 <u>MECHANICAL ISSUES</u>

The staff's review of the licensee's submittal focused on the effects of power uprate on the structural and pressure boundary integrity of the reactor coolant piping, components, their supports, reactor vessel and internal components, the Control Rod Drive Mechanism (CRDM), certain pumps and valves, and the balance-of-plant (BOP) piping systems.

The GE generic guidelines for BWR power uprate were based on a 5% higher steam flow, an operating temperature increase of 5°F and an operating pressure increase of 40 psi or less. For BFN, the maximum reactor vessel dome pressure increases from 1005 psig to 1035 psig (increase 30 psi). The dome temperature increases from 547°F to 551°F (increase 4°F) and the steam flow rate increases approximately 6% from 13.37×10^6 lb_m/hr to 14.15×10^6 lb_m/hr for both Browns Ferry 2 and 3. The maximum core flow rate remains unchanged for the BFN power uprate conditions.

(1) RPV and Internals

The licensee evaluated the reactor vessel and internal components by considering load combinations of the plant licensing basis as specified in the BFN UFSAR. These load

combinations include reactor internal pressure difference (RIPD), LOCA, and seismic and fuel lift loads. The seismic and fuel lift loads are unaffected by the power uprate. The licensee recalculated RIPDs for the power uprate shown in Table 3-2 of Reference 3 for normal, upset and faulted conditions.

The stresses and cumulative fatigue usage factors (CUFs) for the reactor vessel components were evaluated by the licensee, in accordance with the code of record at Browns Ferry, the ASME Code Section III, Subsection NB, 1965 Edition with Summer 1965 addenda for BFN unit 2, and 1965 Edition with Summer 1966 addenda for BFN unit 3. 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 Reference 3. The CUFs for the uprated power level were calculated by using the power uprate scaling factor for limiting components such as feedwater nozzle, recirculation outlet nozzle, main closure stud and support skirt. For these limiting components, the calculated CUFs were provided in Table 3-3 of Reference 3 and the maximum stresses for the uprated power conditions were summarized in the licensee's March 16, 1998 letter, Table 1(c)-2.

The licensee stated that the CUF for the feedwater nozzle was calculated to be 0.984 excluding the inner blend radius location for which the structural integrity was demonstrated separate by the power uprate fracture mechanics calculation in accordance with NUREG-0619. The BFN triple thermal sleeve sparger design at the feedwater nozzle has seal surfaces where there could be potential leakage of the feedwater. The potential leakage can mix the feedwater with the hotter downcomer flow and cause temperature cycling which could yield a significant high cycle fatigue usage. The licensee also indicated that the current BFN NUREG-0619 program includes both the monitoring of seal leakage and the inspection of feedwater nozzle to ensure that the integrity of the nozzle will be maintained for the power uprate condition. The staff finds that the licensee's evaluation is adequate to ensure the structural integrity of feedwater nozzle at the uprated power level.

TVA letter dated March 16, 1998, Table 1-2(c) shows the stress in the support skirt in exceedance of the code limit. In its response dated July 24, 1998, to the staff's RAI, the licensee provided a detailed description and calculation results that demonstrate the reactor vessel support skirt to be acceptable using elastic-plastic analysis in accordance with the ASME Code III Section NB-3228.5. The stresses are in compliance with the Code requirements. The component meets the thermal ratcheting requirement of NB-3222.5 and the CUF for the support skirt was calculated to be less than the Code limit of 1.0. The staff finds the licensee's evaluation to be acceptable.

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

Based on its review, the staff finds that the maximum stresses and fatigue usage factors as provided by the licensee are within the code-allowable limits and concludes that the reactor

vessel and internal components will continue to maintain the structural integrity for the power uprate.

(2) Control Rod Drive Mechanism (CRDM)

The licensee evaluated the adequacy of the CRDM in accordance with the Code of record, the ASME Code Section III, 1968 Edition and Addenda to and including Summer 1970.

In a March 16, 1998 letter, 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 this component results from a maximum CRD internal hydraulic pressure of 1750 psig by an abnormal operating condition. 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.

In addition, the CRDMs have been designed for dome pressure of 1250 psig and operating temperature of 575°F which are higher than the maximum bottom head pressure and temperature of 1070 psig and 532°F, respectively, for the power uprate at BFN.

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

(3) Reactor Coolant Piping and Components

The licensee evaluated the effects of the power uprate conditions, including higher flow rate, higher temperature and higher pressure, 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 licensee indicated that the evaluation was performed using the original Code of record specified in the Browns Ferry UFSAR, 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, reactor vessel bottom head drain line, RWCU, reactor vessel head vent line, RCIC, condensate and feedwater system, CS, HPCI, RHR 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 3-4 and 3-5 of the power uprate license amendment request of Reference 3.

As summarized in Table 3-4 and 3-5 of Reference 3, 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 power uprate, and are, therefore, acceptable. For those systems whose design temperature and pressure did not envelop the uprate power conditions, the licensee reviewed the existing design basis stress analyses and compared the design margins or differences between the calculated stresses and the Code allowable limits against the percentage increases in Tables 3-4 and 3-5. The licensee concluded that the

original analyses of RCPB piping systems have sufficient design margins to accommodate the higher operating flow, pressure and temperature due to the proposed power uprate. 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, appropriate components, connections and supports in a manner similar to the evaluation of the RCPB piping systems based on increases in temperature and pressure of the design basis analysis input. These systems include lines which are affected by the power uprate, but not evaluated in section 3.5 of Reference 3, such as main steam bypass lines, the MSRV discharge, and portions of main steam and feedwater systems outside the primary containment. The licensee reviewed the maximum stress levels and fatigue analysis results and found that there are sufficient margins between actual stresses and Code-allowable limits in the original design to accommodate the specific increases in temperature, pressure and flow rate provided in Tables 3-4 and 3-5. In addition, the licensee indicated that the LOCA and MSRV discharge dynamic loads for the proposed power uprate are within those loads in the existing analysis basis. The piping systems attached to the torus shell are designed for a temperature limit of 177°F which remains the same for the power uprate. The licensee concluded that all piping is below the Code-allowable limits. The staff reviewed the licensee's submittals and finds that the licensee's evaluations are acceptable.

The licensee evaluated pipe supports including anchorages, equipment nozzles, and penetrations by comparing the increased piping interface loads on the system components due to the power uprate thermal expansion, with the margin in the original design basis calculation, and performing detailed analyses using exact load combinations at the uprated conditions. The effect of power uprate conditions on thermal and vibration displacement limits was also evaluated by the licensee for struts, springs and pipe snubbers, and found to be acceptable. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the power uprate, and no new pipe break locations were identified. The staff finds the licensee's evaluation to be acceptable.

Based on the above 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 power uprate.

(4) Equipment Seismic and Dynamic Qualification

The licensee evaluated equipment qualification for the power uprate condition. The dynamic loads such as MSRV 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 Reference 3 because the plant-specific hydrodynamic loads defined during the Mark I Containment LTP for the design-basis analysis at Browns Ferry are bounding for the power uprate.

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

a. Seismic loads are unchanged for the power uprate;
b. No new pipe break locations were identified for the power uprate;
c. MSRV and LOCA dynamic loads, and pipe-whip and jet impingement loads used in the original design basis analyses are bounding for the power uprate.

(5) Safety-Related Safety/Relief Valves and Power-Operated Valves

In its May 22, 1998, response to the staff's RAI dated April 22, 1998, the licensee indicated that the current NRC-approved MSRV setpoint tolerance (+/-3%) was used in the abnormal transient analyses for the power uprate condition. The MSRV setpoints are increased as a result of the reactor dome pressure increase for the power uprate. The analytical limits for the increased MSRV setpoint pressure are provided in Table 5-1 of Reference 3. The licensee determined that peak RPV steam pressure remains below the ASME allowable of 110% of design pressure and that safety-related MSRV operability is not affected by the proposed changes. The licensee stated that the plant-specific analyses for the 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.

The licensee evaluated MSRV air-clearing loads including MSRV discharge line loads, suppression pool boundary pressure loads and drag loads on submerged structures. An increase in MSRV setpoint pressure due to the power uprate will result in higher opening pressure and, therefore, higher MSRV discharge loads. The increased MSRV loads resulting from the increase in the setpoint pressure were compared with BFN plant unique design limits calculated during the MARK 1 LTP. The licensee indicated that the original MSRV load definition used at BFN is more severe than the conditions for the proposed power uprate. Consequently, the licensee concluded that the SRV loads for the SRV discharge line piping will remain unchanged from the original design basis analysis.

Based on its review, the staff concludes that the MSRVs and the MSRV discharge piping will continue to maintain the structural integrity and to provide sufficient overpressure protection to accommodate the proposed power uprate at Browns Ferry.

In a May 22, 1998, letter, the licensee stated that the system evaluations performed to support the BFN Units 2 and 3 power uprate request considered the effects of increased pressure, flow, and temperatures on the functional performance of safety-related pumps and valves. The proposed power uprate will result in a nominal reactor pressure increase of 30 psi and an increase in main steam flow of 6 percent. The increase in nominal reactor pressure was said to result in a 30 psi increase in the lift setpoints for the MSRVs. Ambient room temperature and maximum post-LOCA containment pressure increases due to the power uprate were determined to be about 3 to 5°F and 1.0 psi, respectively. For conservatism, the licensee included slightly higher increases in the assumed nominal reactor pressure and MSRV setpoint.

The licensee reported that plant systems were evaluated to ensure that pump and valve functional requirements continued to be met with the new power uprate conditions. For example, the licensee evaluated MOVs within the scope of its program developed in response to GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The licensee indicated that four GL 89-10 MOVs were identified for each reactor unit as needing torque

switch adjustments for the power uprate conditions which will be accomplished prior to implementation of the power uprate.

As noted in NRC Inspection Report 50-260, 296/98-03, dated June 16, 1998, the licensee has completed its MOV calculations to support the power uprate for Browns Ferry Unit 3. In a letter dated July 24, 1998, the licensee stated that the calculations for Browns Ferry Unit 2 to reflect the power uprate conditions for safety-related pumps and valves will be completed by August 31, 1998. However, the licensee indicated in its July 24, 1998, letter that calculations to support the required modifications due to the power uprate have been completed. In support of the power uprate, the licensee reported, for Unit 2, that four MOVs will require adjustment of their torque switch. For Unit 3, the licensee stated that two MOVs will require torque switch adjustment and that one MOV will require both torque switch adjustment and spring pack replacement. The licensee also indicated that no adjustments or modifications of other power-operated valves have been found to be necessary to support the power uprate. The licensee will implement these modifications prior to the implementation of the power uprate for BFN Units 2 and 3.

In a July 24, 1998, letter, the licensee stated that it has evaluated its response to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," on pressure locking and thermal binding in light of the planned power uprate. The licensee determined that the previous submittal in response to GL 95-07 remains valid without any necessary revision. The licensee also evaluated the increase in the postulated accident containment temperature for the power uprate condition with respect to its response to GL 89-10, GL 95-07, and GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," regarding potential over-pressurization of isolated piping segments at BFN. The licensee evaluated the effect of the temperature increase on MOV motor output, and the potential increase in valve and piping pressure resulting from the higher containment temperature. The licensee did not identify any adverse impact from the postulated increase in containment temperature. The NRC staff will complete the review of the licensee's response to GL 95-07 and GL 96-06 separate from this power uprate request.

Based on its review of the information provided by the licensee, the staff concludes that the requested power uprate will not have an adverse effect on the performance of mechanical components of safety-related pumps and valves at BFN Units 2 and 3.

(6) CONCLUSION

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

13.0 EMERGENCY OPERATING INSTRUCTIONS (EOIs)

For BFN, the EOPs are designated as EOIs. In the letter dated May 20, 1998, TVA stated that review and revision of the EOIs, which includes a review of all EOI variables and limit curves, for changes due to power uprate, is an ongoing activity during power uprate implementation phase. The review, update of the EOIs for any changes, and training are scheduled to be completed prior to the power uprate start-up for Browns Ferry Unit 3. This is acceptable.

14.0 HUMAN FACTORS ISSUES

The staff's evaluation of human factors issues consist of the following five review topics.

Topic 1 - Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures (EOPs and AOPs). Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will new operator actions be required?

By letter dated April 28, 1998, the licensee stated that its EOPs are symptom-based and that changes to the EOPs and AOPs (Emergency Operating Instructions [EOIs] and Abnormal Operating Instructions [AOIs]) consist of revisions to previously defined numerical values (e.g., reactor pressure vessel high pressure scram setpoint value). The type, scope, and nature of the operator actions required for accident mitigation have not changed with no new operator actions being necessary for power uprate.

The staff finds the licensee's response acceptable.

Topic 2 - Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will affect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (reduced/increased) response times. Discuss why any reduced operator response times are needed. Discuss whether any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to complete the required manual actions in the times allowed. Discuss results of simulator observations regarding operator response times for operator actions that are potentially sensitive to power uprate.

In its letter of April 28, 1998, the licensee identified the following operator actions as particularly sensitive to power uprate: initiating emergency RPV depressurization in response to a loss of high-pressure injection; initiating the SLCS following a high-powered ATWS; and inhibiting the ADS during ATWS. The licensee reviewed its PSA and determined operator response times affecting core damage frequency and compared them to those identified in the GE topical report NEDC-31984P, Supplement 2. The GE report evaluated the impact of power uprate on a generic basis. The report indicated that some operator response times would be reduced after power uprate but the effects on operator reliability and performance were determined to be insignificant. The licensee stated that it reviewed all operator responses used in its PSA and confirmed that the effect on operator response times due to power uprate on Browns Ferry was consistent with the GE generic findings.

For the three actions sensitive to power uprate, the licensee compared pre-and post-power uprate times (assumed from the PSA). The licensee indicated that, for the action of recognizing the need to depressurize, which had a preuprate response time of 30 minutes, there would be a postuprate reduction of 2 minutes in operator response time and a 6-second reduction in response time from the 3-5 minute preuprate response time to reach the (-)190 inch vessel level (once the (-)162 level was reached). For the operator action of initiating SLCS given an ATWS with the vessel isolated, under preuprated conditions, the operator had 2 minutes to reach the boron injection initiation temperature (BIIT); under uprated conditions, the operator's response time would be reduced by 6 seconds. For the operator action of inhibiting ADS during ATWS, the preuprate operator response time was 3.5 minutes; the uprate operator response time was reduced by 6 seconds. The licensee stated that, because these operator actions are controlled by EOPs, the slight reduction in response times noted for the power uprate condition "will not significantly affect the operator's ability to safely complete the required actions."

While the staff found that the minimal reduction in response times should not significantly affect the operator's ability to complete the actions, the staff had concern that those actions assumed to be performed in 5 minutes or less by the PSA may be unachievable under realistic conditions. ANSI/ANS Standard 58.8, "Time Response Design Criteria for Safety-Related Operator Actions," 1984), indicates that for events that occur with an estimated frequency of @10^{-1} per reactor year (e.g., reactor coolant system depressurization), a minimum of 5 minutes is allowed for the operator to diagnose the event (i.e., verify automatic responses, observe plant parameters, plan subsequent actions) or the safety-related systems and components "shall be initiated by automatic protection systems" (less frequently occurring events require more diagnosis time). Therefore, the staff requested that the licensee provide supporting information that operators can perform the required tasks under simulated accident conditions in the times assumed by the PSA.

In its July 24, 1998, response, the licensee clarified that "the actions specified to occur within 5 minutes are all in response to an ATWS event." EOI's, the licensee's emergency operating procedures, are used by the operator to complete the steps, "with no other required actions [to] distract the operator from completing the necessary mitigating actions and the accomplishment of the assigned duties." They further state that, for initiating SLCS pump start and inhibiting the ADS, two of the actions of concern, " ... are performed from direct procedural guidance and both can be completed using switches immediately available to reactor operators in the main control room's control area. SLCS pump start is accomplished by turning the SLCS keylock switch (key located next to switch on panel) to either 'Start A' or 'Start B' position. Inhibiting ADS is accomplished by turning two keylock switches (keys located next to switches on panel) to 'Inhibit' position. These are straightforward operator actions and, thus can be performed rapidly." Manually opening individual MSRVs, a third action of concern, is accomplished from the main control room by "placing the control switch to the 'Open' position. This is a straightforward action and, thus, can be performed rapidly." The licensee also clarified that the values for operator actions assumed by the PSA were based on:

> Detailed human action evaluation work [was] accomplished to assign an unique numerical value for selected human actions including the actions associated with dynamic actions accomplished during the plant response to an initiator. These human actions evaluations were accomplished by including an interview with operating personnel [and] guided by selected factors [including]: a) plant man-machine interfaces and indications of conditions;

b) task complexity; c) adequacy of time to accomplish the actions; d) significant preceding and concurrent actions; e) stress; f) training and experience, and ; g) procedural guidance. The interviewees were encouraged to utilize simulator experience and observations as input to the question responses.

The staff finds the licensee's response to the staff concerns acceptable.

The licensee also indicated that one manual action (termination of the HPCI system injection following spurious initiation during an Appendix R fire event covered by their Appendix R Safe Shutdown Instructions (SSIs)) has a reduction in response time of 3 minutes (i.e., from a predicted time of 10 minutes to 7 minutes). This operator action is necessary to prevent flooding of the main steam lines. The licensee explained that the reduction in response time is not a result of the power uprate but rather a result of using different models to predict pre- and post-uprate operator action times (i.e., the GE SAFE model for preuprate predictions and the SAFER model for post-uprate).

The licensee further indicated that this operator action of shutting down HPCI is required in nine Unit 2 SSIs and seven Unit 3 SSIs. One instruction for each Unit requires HPCI shutdown from outside the main control room. The licensee stated that this action (close one valve from the 250V DC reactor motor-operated valve board located just outside the main control room on the same elevation), has been performed in a shorter time than allowed by the SAFER model (i.e., seven minutes).

The staff requested the licensee to explain the statement, on page E1-4 of its letter dated April 28, 1998, "TVA has previously demonstrated that this action, close one valve from the 250V DC reactor motor operated board, located on the same elevation just outside the main control room, can be performed within the shorter time predicted by the SAFER model." The staff asked, when was this demonstration accomplished and how was it accomplished. Were the following factors considered: environmental conditions expected; procedural guidance for the required actions; support personnel and/or equipment required to carry out the required actions; information requirements including qualified instrumentation.

In response to the staff's questions related to manually isolating HPCI, the licensee stated in its July 24, 1998, letter that,

> The Appendix R fire event which requires manual isolation of the HPCI from outside of the control room is a fire in fire zone 16, which is a control building fire in plant elevation 593 though plant elevation 617. ... TVA Operations personnel performed in plant drills on October 25, 1995 to validate all manual operator action times. Drill performance was observed by the NRC and documented by NRC as part of an NRC inspection (NRC Inspection Report 95-60).

The licensee continues by stating that, "For Unit 2, the 10 minute HPCI isolation steps were 33 and 34 of 36 total action steps for the associated Unit 2 procedure. The entire associated procedures performance was completed in 8 minutes, 20 seconds." All actions of the procedure were completed at the control boards in the same Shutdown Board Room, except for travel from the main control room and thus, the licensee indicates that each step of the procedure takes approximately the same amount of time. For the power uprate, the licensee

indicates in its July 24, 1998, letter, that the procedure is being revised, moving the steps which isolate HPCI to the first steps of the procedure. The licensee states that, as a result of the procedure reconfiguration, "this action will be performed in considerably less than 7 minutes."

For Unit 3, the 10 minute isolation steps were steps 10 and 11 of 62 total steps of the associated procedure. Performance of the entire procedure took 12 minutes and 15 seconds. However, all actions of this procedure are not completed in the same Shutdown Board Room, with travel from the Shutdown Board Room to locations ending in the reactor building adding "considerable time not used in performing control switch manipulations for this procedure." However, the licensee states in its July 24, 1998, letter, that the actions to isolate HPCI for Unit 3 "will be performed in considerably less than 7 minutes" and, as with the Unit 2 procedure, the Unit 3 procedure is being revised to move the steps isolating HPCI to the first steps of the procedure. They further describe that the Shutdown Board Room, from which the manual action to isolate HPCI is performed, is protected by a safety-related vent system and is independent from the fire zone. No new controls, special equipment or additional support personnel are required to take the required actions.

The staff finds the licensee's response to the staff concerns acceptable.

The staff also asked for clarification related to the SAFER model referred to by the licensee in its April 28, 1998, response to the staff's RAI, used to analyze LOCA conditions and fuel heatup activities. In its "Power Uprate Safety Analysis for the Browns Ferry Nuclear Plant Units 2 & 3" report (enclosure 5 of the licensee's October 1, 1997, submittal), GE applied the model to analyze an Appendix R fire event and stated that, "Sufficient time is available for the operator to perform the necessary actions" (p.6-9). The staff asked how the SAFER model evaluates human actions and how did GE conclude that operators have sufficient time to perform necessary actions.

The licensee, in its July 24, 1998, letter, clarified that the SAFER model does not evaluate human actions and that the demonstration of operator response times is covered in responses to previous staff questions.

The staff finds the licensee's response to the staff concerns acceptable.

The staff further asked for clarification on the SAFER model being used for the power uprate analysis and the SAFE model for the preuprate analysis. The SAFE model, according to the licensee in its April 28, 1998 submittal, predicted a required operator action time to shut down HPCI of 10 minutes; the SAFER model predicted 7 minutes to shut down HPCI. Which action time is TVA taking credit for?

The licensee stated in its July 24, 1998, letter, that it will implement the 7 minute action time.

The staff finds the licensee's response to the staff concerns acceptable.

Lastly, the staff asked the licensee to address the question of what are credible errors that operators could make in shutting down HPCI (the action as required from the control room and from outside the main control room). What are the consequences of the operator failing to accomplish the action and how will recovery from the failure(s) be accomplished? How does

TVA know that operators can successfully recover from credible errors, i.e., provide evidence that operators can recover from credible errors?

In its July 24, 1998, letter, the licensee explained that "no changes are being made in the design approach of the Appendix R program due to the power uprate. The procedural step changes being made do not introduce any new human factors engineering issues." In addition, operator requalification training with detailed procedures, control board layout and location, "provide a combined influence for a low probability of operator error." The licensee maintains that, in the event of an operator error in the performance of this step, "or any other error affecting reactor control parameters, reactor water level, or reactor pressure, resulting conditions would be recognized by the operators using indications at the backup control panel."

The license indicated that consequences of not making the 7 minute HPCI time action could potentially fill the reactor vessel up to the main steam lines and result in RCIC being disabled by liquid carryover. If the action were not performed within the specified time limits, "recovery actions by the operators would be to drain RCIC and utilize other alternate injection paths in that time frame. High confidence exists that this is a very recoverable scenario... ."

The staff finds the licensee's response to the staff concerns acceptable.

Topic 3 - Discuss all changes the power uprate will have on control room alarms, controls, and displays. For example, will zone markings on meters change (e.g., normal range, marginal range, and out-of-tolerance range)? If changes will occur, discuss how they will be addressed.

In the licensee's April 28, 1998 submittal, it stated that two changes in control room indications will result from the planned power uprate. Three control room indicators (wide range reactor pressure) will be modified to revise the low limit of the indicator scale's "red zone" to accommodate the increase in reactor pressure scram setpoint as a result of the power uprate. Also, forebay temperature indication span will be modified to accommodate a lower forebay high temperature setpoint.

The licensee also indicated that various main control room instructional aids will be revised because of the power uprate. These aids include labels, sketches, or markings which are posted and used as memory aids or instructional guidance. Power/flow map, reactor water level compensation curves at reduced pressure conditions, and main steam relief valve setpoints/locations will also be modified to address the power uprate. All changes will be prepared in accordance with plant modification procedures which includes detailed review of the proposed control room design change package and all required changes will be implemented before operating the plant at uprated power.

The staff finds the licensee's response acceptable.

Topic 4 - Discuss all changes the power uprate will have on the Safety Parameter Display System (SPDS) and how they will be addressed.

The licensee indicated in its April 28, 1998, submittal that the design and intent of the SPDS will remain unchanged. The information presented on the SPDS top-level display and the method of presentation remain unchanged. Changes to SPDS include recalibration of input/output

points; changes to constants which input to the displayed points (e.g., rated core thermal power); changes to MSIV lift setpoints; and changes to EOI limit graphs. The licensee stated that these changes will be transparent to the operator because EOI execution is unaffected.

The staff finds the licensee's response acceptable.

Topic 5 - Describe all changes the 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.

The licensee indicated the following in its April 28, 1998, submittal:

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

The licensee stated that operator training content will be revised to include the power uprate and that classroom and simulator training will be conducted prior to implementation of the power uprate on Unit 3. Additional training will be provided as deemed necessary by the Power Uprate Refueling Test Program. The licensee will compare non-uprate to uprate data sets for major simulator transients.

(b) Complete simulator changes that are consistent with ANSI/ANS 3.5-1985.

The licensee stated that simulator fidelity will be re-validated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." Simulator re-validation 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 that simulator changes will be implemented prior to installation of power uprate modifications in the plant. Two software loads will be maintained, one for power uprate changes and the other for Unit 2 configuration and non-uprate conditions. Differences will be recorded in a Unit 2/Unit 3 Differences listing, which will be incorporated in the training program. Simulator re-validation in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, will be accomplished in two stages. First, the simulator performance will be validated against the power uprate design and engineering analysis data. Second, once plant modifications are completed on Unit 3, data will be collected and compared with simulator performance data. TVA intends to integrate this re-validation with the BFN simulator quadrennial testing and certification program.

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

See evaluation of Topics 3 and 4, above.

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

The licensee stated that the simulator changes will be integrated with the BFN simulator quadrennial testing and certification program. This program is understood to provide administrative controls, practices, and procedures sufficient to reflect issues and discrepancies identified during startup.

In its April 28, 1998, letter, the licensee made the following commitments with respect to simulator modifications, and classroom and simulator training:

1. Operator training content will be revised to include power uprate. Classroom and simulator training will be conducted prior to restart of Unit 3 at uprated conditions.

2. The plant simulator control room and plant simulator process computer changes will be completed as required to support the training program. TVA expects to complete this action prior to simulator training for power uprate. By letter dated September 1, 1998, TVA stated that this action has been completed.

3. An acceptance test will be run to benchmark simulator performance based on design and engineering analysis data. The primary focus will be matching the heat balance data for the 105 percent power steady-state test as required in ANSI/ANS 3.5-1985, Appendix B1.1. TVA expects to complete this action prior to simulator training for power uprate.

4. Once the power uprate modifications are completed on Unit 3, data will be collected, including the post power uprate refueling test report, and compared to simulator data as required by ANSI/ANS 3.5-1985, Section 5.4.1. TVA expects to complete this as part of BFN's required simulator certification program. TVA expects to submit this report within 60 days following completion of certification.

5. Operator training will be modified, as necessary, based on observations of plant and operator performance during the Power Uprate Refueling Testing Program.

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 power uprate. The staff finds the licensee's response acceptable. These commitments, as identified in Items (1) through (5), above, have been incorporated into the Facility Operating Licenses for both units as additional License Conditions.

Based on the above discussion, the staff concludes that (a) the analyses and evaluations supporting the power uprate follow the generic guidelines approved by the staff and hence is acceptable and (b) the power uprate for the BFN units 2 and 3 should not adversely affect operator actions or operator reliability.

(15) <u>TEST CONTROL</u>

(1) Regulatory Basis

Regulatory provisions for the testing of structures, systems, and components are identified under Criterion XI, "Test Control," of Appendix B to 10 CFR 50. The program for implementing these requirements is described in section 9.4 of the TVA Nuclear Quality Assurance Plan

(NQAP), Reference 6. The NQAP description follows the guidance of ANSI N45.2-1971, section 12 (Reference 7) with respect to the development of test procedures, conduct of testing, and documentation and evaluation of test results. Test results are documented in a suitable test package, including deviations and adverse conditions, and actions taken to resolve the condition. Documentation of test results are maintained as lifelong records, in accordance with the guidelines provided by Regulatory Guide 1:28, Revision 3 (Reference 8), to which the licensee has committed.

(2) Generic Test Guidelines for GE BWR Power Uprate

NEDO-31984, section 5.11.9 provides the general guidelines for power uprate testing.

> A testing plan will be included in the uprate licensing application. It will include pre-operational tests for systems or components which have revised performance requirements. It will also contain a power increase test plan.

Guidelines to be applied during the approach to and demonstration of uprated operating conditions are provided in section L.2 of GE proprietary report NEDC-31897P-1. GE report NEDC-32751P, submitted with the licensee's application, provides the required additional information relative to power uprate testing.

(3) Startup Test Plan

The license will conduct limited startup testing at the time of implementation of power uprate. The tests will be conducted in accordance with the guidelines of NEDC-31897P-1 to demonstrate the capability of plant systems to perform their designed functions under uprated conditions.

The tests will be similar to some of the original startup tests, described in section 13.5 of the licensee's UFSAR. Testing will be conducted with established controls and procedures, which have been revised to reflect the uprated conditions. Revised plant procedures, reflecting the uprate conditions, will be used to the extent practicable during the test program.

The tests consist essentially of steady state, baseline testing between 90 and 100 percent of the currently licensed power level. At least one set of data will be obtained between 100 and 103 percent power, and a final set of data at the uprated (105%) power level. The tests will be conducted in accordance with existing site procedure SSP 8.1, "Conduct of Testing," which implements the licensee's QA program test control requirements.

The following power increase test plan is provided in NEDC-32751P, section 10.5, "Required Testing."

(a) Surveillance testing will be performed on the instrumentation that requires recalibration for power update.

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

(c) Power increases beyond the previous rating will be made in increments along an established flow control/rod line. Steady state operating data, including fuel thermal margin, will be taken and evaluated at each step.

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

The licensee's test plan follows the guidelines of NEDC-31897P-1 and the staff position regarding individual power uprate amendment requests.

(4) Performance Testing

(a) Systems/Components with Revised Performance Requirements

NEDO-31897, section 5.11.9 guidelines specify that pre-operational tests will be performed for systems or components which have revised performance requirements. The licensee's submittal, NEDO-32751P, identifies the following systems and components that have revised performance requirements.

- Reactor Recirculation System (RRS) - The RRS system operating pressure, temperature, drive motor horsepower, pump flow, and pump brake horsepower will be increased.

  These tests will demonstrate flow control over the entire pump speed range to enable a complete calibration of the flow control instrumentation. The tests will also demonstrate acceptable reactor internal vibration levels at the uprated conditions.

- RCIC - The RCIC pump maximum specified pump and turbine rated speed will increase in order to deliver its design flow rate at the higher pump discharge pressure.

- HPCI - The HPCI pump maximum specified pump and turbine rated speed will increase in order to deliver its design flow at the higher pump discharge pressure.

  RCIC and HPCI tests will demonstrate that they are capable of meeting flow requirements at the increase pump discharge pressures. These tests will also demonstrate that the reliability of these systems are not decreased by the higher loads placed on the systems or because of any system modifications to compensate for the increased loads. Since theses systems are within the scope of the Maintenance Rule, system reliabilities will be established and tracked.

- Instrument setpoints (e.g., high pressure scram) will be established based on the power uprate safety analyses. New analytic limits for setpoints, including reactor high pressure trip and ATWS high pressure recirculation pump trip will be established.

- Pressure Control System (PCS) - The PCS consists of the pressure regulation system, turbine control valve system, and steam bypass valve system. Power uprate testing will confirm that the pressure regulating tuning parameters set prior to startup are acceptable, or test results will indicate what tuning parameter adjustments are necessary.

- Turbine Generator - The increased reactor operating pressure associated with uprated conditions has the potential to result in increased turbine overspeeding during system startup, increasing the probability of the system to trip during startup.

- SLCS - The pressure at which the system injects will be increased.

- MSRV - The MSRV set point pressure will be increased.

- EDGs - Some additional loads may be added.

- MOVs - Pursuant to the guidance of GL 89-10, modifications may be necessary to some motor operated-valves.

(b) Planned Performance Tests

The licensee plans to conduct tests during the ascension to power uprate conditions. The performance tests and associated acceptance criteria are based on the Browns Ferry original startup test specifications and previous GE BWR power uprate test programs. The licensee has identified performance tests for the following systems:

- Reactor Recirculation
- RCIC
- HPCI
- Electro-Hydraulic Pressure Control
- Turbine Generator
- SLCS
- Reactor Water Level Measurement
- Feedwater Control
- Drywell Atmosphere Cooling
- Intermediate Range Neutron Monitoring
- Average Power Range Monitoring

With regard to systems and components with revised performance requirements identified in the preceding section, but not identified above:

- MSRVs will be reset at Wyle Laboratories in accordance with the inservice inspection program (see revised surveillance requirement SR 3.4.3.1).

- EDG - The power uprate analysis, performed subsequent to the licensee's submittal, concluded that sufficient margin existed in diesel generator capacity that the additional electrical loading would have an insignificant impact on the station load flows. (See licensee's letter dated May 20, 1998, Section B.2). EDG testing conducted under SR 3.8.1.7, provides assurance that the diesels can fulfill their design requirements; no special testing is planned as part of the startup test program.

- MOV testing - Post-modification testing of motor-operated valves modified as a result of the power upgrade will be conducted as part of the design change control closure process; no special testing is planned as part of the startup test program.

The planned performed tests follow the guidelines of NEDC-31897P-1 with respect to performance testing of systems or components that have revised performance requirements.

(5) Conclusions

The licensee's program for startup testing follows the guidelines of NEDC-31897P-1, which has been accepted by the NRC as the generic basis for power uprate amendment requests. The submittal provides a test program that follows NEDC-31897P-1 guidelines for uprate testing and meets 10 CFR 50, Appendix B requirements for test control. The licensee's power uprate test program is acceptable.

16.0 EVALUATION OF CHANGES TO FACILITY OPERATING LICENSE AND TS

As a result of the power uprate, the licensee proposed following TS changes:

(1) TS 1.1, Definitions

Revise the RTP value from 3293 to 3458 MWt to reflect increased power level.

(2) SR 3.1.7.6, and TSB 3.1-44

Revise the value of SLCS pump discharge pressure from 1275 to 1325 psig to account for the effects of increased main steam relief valve setpoints.

(3) Table 3.3.1.1- 1, Function 2.b

Revise the allowable value for APRM Flow Biased Simulated Thermal Power - High Scram setpoint intercept from 71 percent to 66 percent RTP to maintain the preuprate power/flow relationship in terms of absolute power.

(4) Table 3.3.1.1- 1, Function 3

Revise the allowable value for Reactor Vessel Steam Dome Pressure - High scram setpoint from 1055 to 1090 psig to account for increased 30 psi reactor operating pressure.

(5) SR 3.3.4.2.3b

Revise the allowable value for Reactor Steam Dome Pressure - High Anticipated Transient Without Scram-Recirculation Pump Trip (ATWS RPT) setpoint from 1146.5 to 1175.0 psig to account for increased 30 psi reactor operating pressure.

(6) (For Unit 2 Only) Table 3.3.6.1-1, Function 5

Revise the allowable value for RWCU system isolation to 188°F

(7) Figure 3.4.1-1, TSB 3.4-3, Applicable Safety Analyses

Revise lower bounds (rod lines) for Thermal Power vs Core Flow Stability Regions from 108%, 100% and 80% to 102.9%, 95.2% and 76.2% respectively, to maintain the preuprate power/flow ratios in terms of absolute power.

(8) SR 3. 4. 3. 1

Revise the MSRV setpoints from 1105, 1115 and 1125 to 1135, 1145 and 1155 respectively, to account for increased 30 psi reactor operating pressure.

(9) LCO 3.4.10, SR 3.4.10.1

Revise the LCO and SR for reactor steam dome pressure to be less than 1050 psig.

(10) SR 3.5.1.7, TSB 3.5-12, SR 3.5.1.7, SR 3.5.3.3, TSB 3.5-28, SR 3.5.3.3

Revise upper and lower bounds of HPCI, and RCIC pump test pressure from 1010 and 920 psig to 1040 and 950 psig, respectively, to account for increased 30 psi reactor operating pressure.

(11) LCO 3.7. 1, Condition D, SR 3.7.1.2 p. 3.7-3a, Figure 3.7.1-1 p. 3.7-5, SR 3.7.2.1 TSB 3.7-1, Background, TSB 3.7-2, Applicable Safety Analyses, LCO, Applicability p. B 3.7-5, Actions, TSB 3.7-8, LCO

Add LCO and Action statement addressing UHS for RHRSW. Add upper limit for the UHS average water temperature for RHRSW to be in accordance with new Figure 3.7.1-1. The value is used for calculating long-term suppression pool temperature response in the new containment analysis. Add a note to for the UHS average water temperature for ECCW to refer to additional UHS requirements from RHRSW.

TSB 3. 5-4, Background, TSB 3.5-24, Background

Revise value of upper design pressure for HPCI and RCIC operating range 54 psi from 1120 to 1174 psig to account for the effects of increased main steam relief valve setpoints.

TSB 3. 6-2, TSB B 3. 6-7, Applicable Safety Analyses

Revise value of P., (from 49.6 to 50.6 psig) to account for results of the new containment analysis.

TSB 3.7-2 B 3.7.1 B 3.7-6, References

Revise flow rate for RHRSW (from 4.500 to 4000 GPM) used in peak containment temperature analysis. Revise peak containment pressure (from 49.6 to 50.6 psig) to reflect new analysis results. Revise reference for peak suppression pool temperature.

As discussed in Section 2.0 of this SE, TVA's planned approach for achieving the higher power 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 the feedwater flow, (3) no increase in the maximum core flow, (4) a small (less than 3 percent) increase in the reactor operating pressure, and (5) reactor operation primarily along extensions of pre-uprated rod/flow control lines. The operating pressure will be increased approximately 30 psi to assure satisfactory pressure control and pressure drop characteristics for the increased steam flow. For the reasons discussed in Sections 2 through 16 of this SE, the staff finds the above system design and performance conditions will not adversely impact the safe operation of the plant at the increased power level, and therefore, acceptable. The staff also finds reasonable assurance that the radiological consequences of normal operations and anticipated accidents at the BFN Units 2 and 3 will continue to meet regulatory requirements at a rated thermal power of 3458 megawatts and, therefore, are acceptable. The staff has reviewed the licensee's proposed TS changes and determined that they reflect operation at the increased power level, and therefore, acceptable.

17.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official (Kirk Whatley) was notified of the proposed issuance of the amendment. The State official had no comments.

18.0 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 September 1, 1998, (63FR46491).

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

19.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

20.0 REFERENCES

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

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

3. GE Nuclear Energy, "Power Uprate Safety Analysis for the Browns Ferry Nuclear Plant Units 2 & 3," Licensing Topical Report NEDC32751P, Class III (Proprietary), September 1997.

4. GE Nuclear Energy, "Browns Ferry Nuclear Plants Units 1, 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (Revision 1)," NEDC-32484P, February 1996.

5. SECY-91-401, "Generic Boiling Water Reactor Power Uprate Program," December 12, 1991.

6. TVA Quality Assurance Plan (TVA-NQA-PLN89-A), Revision 7, submitted by letter dated October 11, 1996.

7. ANSI N45.2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants," October 1971.

8. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction," Revision 3, 1985.

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