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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 254
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment by Tennessee Valley Authority (the licensee), 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, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 2.C.(1) and 2.(C). 2 of Facility Operating License No. DPR-52 are hereby amended to read as follows:

(1) Maximum Power level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 254, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 254, are hereby incorporated into this license. Tennessee Valley Authority shall operate the facility in accordance with the Additional Conditions.

4. This license amendment is effective as of its date of issuance and shall be implemented before Cycle 11 operation.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment 1: Page 3 of License DPR-52
Attachment 2: Appendix B
Attachment 3: Changes to the Technical Specifications

Date of Issuance: September 8, 1998

(2) Pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use at any time source and special nuclear material as reactor fuel in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 254 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 253 to Final Operating License DPR-52, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment. For SRs that existed prior to Amendment 253, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment.

APPENDIX B

ADDITIONAL CONDITIONS

Amend. Number	Additional Conditions	Implementation Date
253	The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's application dated September 6, 1996, as supplemented May 1, August 14, November 5 and 14, December 3, 4, 11, 22, 23, 29 and 30, 1997, January 23, March 12, April 16, 20 and 28, May 7, 14, 19 and 27, and June 2, 5, 10 and 19, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.	This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.
254	TVA will perform an analysis of the design basis loss-of-coolant accident to confirm compliance with General Design Criterion (GDC)-19 and offsite 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 its 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 to control room operators to maintain doses within GDC-19 guidelines. This ability will be maintained until the required modifications, if any, are complete.	This amendment is effective immediately.

APPENDIX B

ADDITIONAL CONDITIONS

254	Classroom and simulator training on all power uprate related changes that affect operator performance will be conducted prior to operating at uprated conditions. Simulator changes that are consistent with power uprate conditions will be made and simulator fidelity will be validated in accordance with ANSI/ANS 3.5-1985. Training and the plant simulator will be modified, as necessary, to incorporate changes identified during startup testing.	This amendment is effective immediately.

ATTACHMENT TO LICENSE AMENDMENT NO. 254

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

Remove	Insert
1.1-6	1.1-6
3.1-25	3.1-25
3.3-7	3.3-7
3.3-8	3.3-8
3.3-35	3.3-35
3.3-61	3.3-61
3.4-4	3.4-4
3.4-8	3.4-8
3.4-30	3.4-30
3.4-31	3.4-31
3.5-6	3.5-6
3.5-13	3.5-13
3.7-1	3.7-1
3.7-2	3.7-2
3.7-3	3.7-3
3.7-4	3.7-4
3.7-5	3.7-5
3.7-6	3.7-6
3.7-7	3.7-7
3.7-8	3.7-8
3.7-9	3.7-9
3.7-10	3.7-10
3.7-11	3.7-11
3.7-12	3.7-12
3.7-13	3.7-13
3.7-14	3.7-14
3.7-15	3.7-15
3.7-16	3.7-16
3.7-17	3.7-17
3.7-18	3.7-18
3.7-19	3.7-19
--------	3.7-20
5.0-20	5.0-20

Remove	Insert
B3.1-54	B3.1-54
B3.4-5	B3.4-5
B3.4-67	B3.4-67
B3.4-68	B3.4-68
B3.4-69	B3.4-69
B3.5-5	B3.5-5
B3.5-16	B3.5-16
B3.5-31	B3.5-31
B3.5-35	B3.5-35
B3.6-3	B3.6-3
B3.6-8	B3.6-8
B3.6-37	B3.6-37
B3.7-1	B3.7-1
B3.7-2	B3.7-2
B3.7-3	B3.7-3
B3.7-4	B3.7-4
B3.7-5	B3.7-5
B3.7-6	B3.7-6
B3.7-7	B3.7-7
B3.7-8	B3.7-8
B3.7-9	B3.7-9
B3.7-1 0	B3.7-1 0
B3.7-13	B3.7-13

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 conside ·ed 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_{NOT} 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 overspeed, 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 ENGINEERED SAFETY FEATURES

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 uprate 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.

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 tc 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.