March 6, 2007

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING FIVE PERCENT UPRATE (TAC NO. MD3048) (TS-431)

Dear Mr. Singer:

The Commission has issued the enclosed Amendment No. 269 to Renewed Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1. This amendment is in response to an application dated September 22, 2006, which supplements a June 28, 2004, application for an increase in licensed thermal power from 3293 megawatt thermal (MWt) to 3458 MWt. This represents an approximate 5-percent increase above the original licensed thermal power (OLTP) of 3293 MWt.

The amendment also changes the Unit 1 licensing bases and associated Technical Specifications to credit 3 pounds per square inch gauge (psig) for containment accident pressure following a loss-of-coolant accident and increase the reactor steam dome pressure by 30 psig. As indicated in a letter dated September 27, 2006, the U.S. Nuclear Regulatory Commission staff Safety Evaluation contains a License Condition specifying that within 30 days of reaching 105-OLTP, large transient testing will be performed, which includes a turbine generator load reject and a main steam isolation valve closure with valve position scram. Another License Condition specifies satisfactory completion of condensate booster pump, condensate pump, and feedwater pump trip testing at 105-percent OLTP.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

L. Raghavan, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-259

Enclosures:
1. Amendment No. 269 to DPR-33
2. Safety Evaluation

cc w/enclosures: See next page

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cc w/enclosures: See next page

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Date: March 6, 2007

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 269
Renewed License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (TVA, the licensee) dated June 28, 2004, as supplemented on September 22, 2006, 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 Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 269, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the Operating License is amended as indicated in the attachment to this license amendment and subject to the following License Conditions:

   During the power uprate power ascension test program and prior to exceeding 30 days of plant operation above a nominal 3293 megawatts thermal power level (100-percent original licensed thermal power (OLTP)) or within 30 days of satisfactory completion of steam dryer monitoring and testing that is necessary in order to achieve 105-percent OLTP (whichever is longer), with plant conditions stabilized at 105-percent OLTP, TVA shall perform a main steam isolation valve closure test and a turbine generator load reject test. Following each test, TVA shall confirm that plant response to the transient is as expected in accordance with previously established acceptance criteria. The evaluation of the test results for each test shall be completed, and all discrepancies resolved, prior to resumption of power operation.

   During the power uprate power ascension test program and prior to exceeding 30 days of plant operation above a nominal 3293 megawatts thermal power level (100-percent OLTP) or within 30 days of satisfactory completion of steam dryer monitoring and testing that is necessary for achieving 105-percent OLTP (whichever is longer), with plant conditions stabilized at 105-percent OLTP, TVA shall trip a condensate booster pump, a condensate pump, and a main feedwater pump on an individual basis (i.e., one at a time). Following each pump trip, TVA shall confirm that plant response to the transient is as expected in accordance with previously established acceptance criteria. Evaluation of the test results for each test shall be completed and all discrepancies resolved in accordance with corrective action program requirements and the provisions of the power ascension test program.

4. This license amendment is effective as of its date of issuance and shall be implemented prior to the restart of Unit 1.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Catherine Haney, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: March 6, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 269

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Replace Page 3 of Renewed Operating License DPR-33 with the attached Pages 3, 5 and 6.

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached 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
3.3-6	3.3-6
3.3-7	3.3-7
3.3-34	3.3-34
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
5.0-20	5.0-20

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 269

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

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TO RENEWED FACILITY OPERATING LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

1.0 INTRODUCTION

1.1 Application

By letter dated June 28, 2004, the Tennessee Valley Authority (TVA, the licensee), submitted an amendment request for Browns Ferry Nuclear Plant (BFN), Unit 1, as supplemented by letters dated August 23, 2004, February 23, April 25, June 6, and December 19, 2005, February 1 and 28, March 7, 9, 23, and 31, April 13, May 5 and 11, June 12, 15, 23 and 27, and July 6, 21, 24, 26, and 31, and August 4 and 18, and September 1, 15 and 22, October 3, 5, 13, and November 6, 2006. The September 22, 2006, supplement requested interim approval of an increase in licensed thermal power from 3293 megawatt thermal (MWt) to 3458 MWt. This represents an approximate 5-percent increase above the original licensed thermal power (OLTP) of 3293 MWt. The initial June 28, 2004, application requested a 20-percent increase above the OLTP. The U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's review of this initial application will be addressed in a separate correspondence.

The proposed amendment would also change the Unit 1 licensing bases and associated Technical Specifications (TSs) to credit 3 pounds per square inch gauge (psig) for containment accident pressure following a loss-of-coolant accident (LOCA) and increase the reactor steam dome pressure by 30 psig.

1.2 Background

Unit 1 is a boiling-water reactor (BWR) plant of the BWR/4 design with a Mark-1 containment. Unit 1 is one of three BWR/4 units at the Decatur, Alabama site. The NRC originally licensed Unit 1 on December 20, 1973, for operation at 3293 MWt.

The construction permit for Unit 1 was issued by the Atomic Energy Commission (AEC) on May 10, 1967. The plant was designed and constructed based on the proposed General Design Criteria (GDC) published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (draft GDC). The AEC published the final rule that added Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, General Design Criteria for Nuclear Power Plants, in the *Federal Register* (36 FR 3255) on February 20, 1971 (GDC).

Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. As discussed in the NRC Staff Requirements Memorandum for SECY-92-223, dated September 18, 1992 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971. At the time of promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the final GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. Each plant licensed before the final GDC were formally adopted, was evaluated on a plant-specific basis, determined to be safe, and licensed by the Commission.

As discussed in Appendix A of the Updated Final Safety Analysis Report (UFSAR), the licensee has made changes to the facility over the life of the plant that may have invoked the final GDC. The extent to which the final GDC have been invoked can be found in specific sections of the UFSAR and in other design and licensing basis documentation. During the construction permit licensing process, Unit 1 was evaluated against the 27 then-current draft of the AEC Proposed GDC. Although neither version of these proposed criteria had been adopted as regulatory requirements, the design, material procurement, and fabrication of each reactor unit were responsive to the respective applicable criteria for a construction permit. Although the later criteria (AEC-70) did not wholly complement the earlier (AEC-27), and also contained many aspects which could have been modified or clarified before their formal adoption, the design bases of each unit of this plant were reevaluated, at the time of initial FSAR preparation against the draft of the 70 criteria current at the time of operating license application.

By design, various systems are cross-tied to support multi-unit operation. For example, ‘A’ loop of the Unit 2 Residual Heat Removal (RHR) System is available to support the RHR system for Unit 1 and vice versa, if needed. For the Control Rod Drive (CRD) systems (CRDS), there is a swing pump between the Units 1 and 2 in the event either of the main pumps is removed from service due to maintenance or a problem.

For Secondary Containment Ventilation, the refuel and reactor zones are common to all three units so the isolation function must remain operable. This functionality also supports the operation of standby gas treatment which provides an emergency filtered and elevated release of secondary containment atmosphere in the event of an airborne contamination condition. In this situation, the normal ventilation is required to be isolated. Also, the reactor building and refuel floor radiation monitors are maintained operable for detection of radiological conditions which would require the isolation of normal ventilation and actuation of the standby ventilation system. These radiation monitors also require that parts of the primary containment isolation system, as well as the reactor protection system (RPS), are maintained operable.

The Unit 1 standby alternating current (ac) electrical boards support the control room emergency ventilation system and control bay chiller and also contain electrical supply boards to support the RPS. Units 1 and 2 have four common emergency diesel generators (EDGs) that are maintained operable to supply emergency electrical power. The Unit 1 station service electrical transformers are necessary for maintaining the operability of the common system.

Equipment necessary to perform Emergency Operating Instructions (EOIs), such as the Unit 1B CRDS pump (which can be used to supply water to Unit 2) and the standby liquid control (SLC) system (SLCS) boron tank concentration (in the event that the boron is needed to supply the other units), are maintained operable.

1.3 Licensee's Approach

The licensee's application for this proposed uprate follows the guidance in the Office of Nuclear Reactor Regulation's (NRR's) Review Standard (RS)-001, Review Standard for Extended Power Uprates (RS-001), to the extent that the review standard is consistent with the design basis of the plant. The technical bases for this request follow the guidelines contained in GE [General Electric] LTR [Licensing Topical Reports] for Extended Power Uprate Safety Analysis NEDC-32424P-A, *Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate* (ELTR1) approved by the NRC in a letter dated February 8, 1996, and NEDC-32523P-A, *Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate* (ELTR2) approved by the NRC in a letter dated September 14, 1998.

The licensee initially intended to perform an extended power uprate (EPU) to 120-percent OLTP, however as information needed to support the EPU review was not available, the licensee supplemented the submittal by requesting a two-step approach. The licensee requested an interim approval of 5 percent, with a subsequent approval to increase to the full 20 percent upon submittal of needed documentation to allow the NRC staff to complete the 20-percent review. Where possible, the licensee used the analyses performed at 20-percent uprate to support the 5-percent uprate submittal.

Enclosure 4 of the June 28, 2004, submittal contains GE NEDC-33101P, *Browns Ferry Unit 1 Safety Analysis Report for Extended Power Uprate*, Revision 0, hereafter referred to as the PUSAR. This document is TVA's assessment of GE's Evaluations for EPU and provides a description of the reviews, audits, and independent verifications preformed by TVA to support the EPU Safety Assessment. Enclosure 5 contains a background, description, and justification of the requested change to the Unit 1 licensing bases to include credit for Containment Overpressure following design-basis accidents (DBAs). Enclosure 7 provides a listing of planned modifications for EPU implementation. Enclosure 8 provides the Unit 1 EPU startup test program, which supplements PUSAR Section 10.4. Enclosure 11 provides a summary of the Grid Adequacy and Stability Study for BFN.

For the Unit 1 uprate to 105-percent OLTP, a higher steam and feedwater (FW) flow is achieved by increasing the reactor power along specified control rod and core flow lines, and increasing reactor operating pressure approximately 30 psig. This increase in steam flow will enable increasing the electrical output of the plant. Additionally, adequate Net Positive Suction Head (NPSH) margin is required during an LOCA to assure proper pump operation. TVA's NPSH evaluation shows that Unit 1 requires containment overpressure credit to ensure adequate NPSH for the emergency core cooling system (ECCS) pumps; therefore, the licensee is requesting approval for containment overpressure credit of 3 psig for the duration of the accident.

The scope of the staff's review included "lessons learned" from past power uprate amendment reviews. In reviewing the licensee's request, the NRC staff considered the recommendations of the report of the Maine Yankee Lessons Learned Task Group (SECY-97-042, *Response to OIG [Office of the Inspector General] Event Inquiry 96-04S Regarding Maine Yankee*, February 18, 1997). The task group's main findings centered on the use and applicability of the computer codes and analytical methodologies used for power uprate evaluations. The NRC staff requested that the licensee identify all codes and methodologies used to obtain safety limits and operating limits and explain how they verified these limits were correct for the uprated core. The licensee was also requested to identify and discuss any limitations imposed by the staff on the use of these codes and methodologies.

1.4 Plant Modifications

All three units were voluntarily shut down by the TVA in March 1985 to address performance and management issues. Following the shutdowns, TVA specified corrective actions which would be completed prior to restart and confirmed their commitment not to restart any unit without NRC's concurrence. All three units retained their operating licenses during their respective long-term shutdown. Unit 2 restarted in May 1991, and Unit 3 restarted in November 1995, following Commission briefings and NRC Staff approval. Prior to the restart of these units, the NRC completed significant inspections and closely monitored restart activities to assure that TVA had adequately corrected the issues that caused the shutdown of all three units.

TVA has maintained Unit 1 in a defueled lay-up condition since 1985. TVA is implementing programs on Unit 1 that are similar to those used for the restart of Units 2 and 3, incorporating improvements and lessons learned. The primary difference between the Unit 1 restart and the previous restart efforts for Units 2 and 3 is TVA's approach to remove and replace large piping sections and components, such as the recirculation piping, rather than analyze and attempt to recover them.

The licensee has determined that several plant modifications are necessary to implement the proposed uprate. Some of these installations/replacements are in anticipation of approval of an uprate to 120-percent OLTP, while others are needed to support restart. The following is a list of some of the major modifications, and a more complete list was provided in TVA's letter dated February 25, 2005.

- Turbine - Replace high and low pressure rotors;
- Condensate and Condensate Booster Pumps - Replace impellers and install new motors;
- FW Pumps - Replace pumps and turbine rotor;
- FW Heaters (FWH) - Replace nozzles and relief valves for FWH 1, 2 and 3;
- Main Generator - Rewind generator stator and field, re-rate generator;
- Main Bank Transformers (MBT) - Install transformers;
- Main Steam Isolation Valve (MSIV) - Replace MSIV poppets and install MSIV stems;
- Recirculation Pumps - Re-rate pumps and motors;
- Jet Pumps - Install sensing line clamps;

- Local Power Range Monitors (LPRMs) - Replace the LPRMs;
- (MS) Relief Valves (MSRV) - Increase mechanical setpoint and install pressure actuation logic; and
- Structural - Modify supports, snubbers, steel beams, and connections.

In addition the licensee has also proposed to install/replace:

- Piping - Installing/replace approximately 16,365 ft of large bore and 27,630 ft of small bore piping;
- Hangars - Install/replace approximately 1,745 ft of large bore and 6,130 ft of small bore piping;
- Conduit - Install/replace approximately 162,150 ft of conduit and 19,300 ft of conduit supports; and
- Cable - Install/replace 844,250 ft of cable.

The NRC staff's evaluation of the licensee's proposed plant modifications is provided in Section 2.0 of this safety evaluation (SE).

1.5 Method of NRC Staff Review

The NRC staff's review of the Unit 1 uprate application is based on RS-001. The RS-001 contains guidance for evaluating each area of review in the application, including the specific GDC used as the NRC's acceptance criteria. The guidance in RS-001 is based on the final GDC found in 10 CFR 50 Appendix A. By application dated February 23, 2005, TVA submitted a supplement to the EPU which provided a matrix that cross-references the draft GDC to the final GDC. It also contained a revision to the template SE in RS-001 replacing the numeric values of the final GDC with the corresponding TVA design criteria and draft GDC that constitute the current licensing basis. Related changes to TVA plant-specific design criteria were also incorporated in the revised template. Minor changes to the template were provided in an additional supplement dated March 7, 2006.

The NRC staff's review evaluates the licensee's assessment of the impact of the proposed uprate on design-basis analyses. As a result of the sequence of submittal of information (120-percent submittal in June 2004, then 105 percent in September 2006), the NRC staff relied on information from the EPU application. The NRC staff also performed audits of analyses supporting the interim uprate and the EPU and performed independent calculations, analyses, and evaluations as noted below.

To the extent practical, the NRC staff has considered the experience from Units 2 and 3 (operating units) and other plants of the same design. Therefore, for program and procedures, the NRC staff considered the history of those programs used on the operating units. The NRC staff also considered the guidance provided in ELTR1, ELTR2, and the supplements.

The NRC staff safety conclusions with regard to reactor core related technical areas for power operation are based on either the generic assessment or plant-specific evaluation. For some items, bounding analyses and evaluations provided in ELTR2 were cited. The ELTR2 generic evaluations assume (a) up to a 20-percent increase in the thermal power; (b) an increase in operating dome pressure up to 1095 psi absolute (psia); (c) a reactor coolant temperature increase to 556 degrees F; and (d) a steam and FW flow increase of about 24 percent. The scope of the NRC staff's review included "lessons learned" from past power uprate amendment reviews.

In areas where the licensee and its contractors used NRC-approved or widely accepted methods in performing analyses related to the proposed power uprate, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. In addition, the NRC staff considered the effects of the changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the proposed uprate conditions.

Audits of the analyses supporting the proposed power uprate were conducted by the NRC staff and its contractors in relation to the following topics:

- information provided in the EPU application for the interim uprate; portions of these audits are applicable for this review (Section 2.0),
- long-term stability solution (LTSS) implementation (Section 2.8.3); and
- anticipated transients without scram (ATWS) instability (Section 2.8.5.7)

Independent confirmatory calculations, analyses, and evaluations were performed by the NRC staff for the following topics:

- peak clad temperatures associated with small-break and large-break LOCAs (Section 2.8);
- percent drop in upper shelf energy (USE) values for the limiting plate and welds for the reactor vessel (RV) (Section 2.1.2);

The results of the calculations are discussed in section 2.8 of this SE.

2.0 EVALUATION

2.1 Materials and Chemical Engineering

The reactor vessel (RV) material surveillance program provides a means for determining and monitoring the fracture toughness of the RV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RV. The NRC staff's review primarily focused on the effects of the proposed uprate on the licensee's RV surveillance capsule withdrawal schedule.

2.1.1 Reactor Vessel Material Surveillance Program

The NRC's acceptance criteria are based on (1) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed and constructed to have an exceedingly low probability of gross rupture or significant leakage; (2) draft GDC-33, insofar as it requires that the RCPB be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant; (3) draft GDC-34, insofar as it requires that the RCPB be designed to minimize the probability of rapidly-propagating-type failures; (4) 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the RV beltline region; and (5) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in NUREG-0800, *Review of Safety Analysis Reports for Nuclear Power Plants* (SRP) Section 5.3.1 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

The NRC's regulatory requirements related to the establishment and implementation of a facility's RV materials surveillance program and surveillance capsule withdrawal schedule are given in 10 CFR Part 50, Appendix H. Two specific alternatives are provided with regard to the design of a facility's RV surveillance program, which may be used to address the requirements of Appendix H to 10 CFR Part 50.

The first alternative is the implementation of a plant-specific RV surveillance program consistent with the requirements of American Society for Testing and Materials (ASTM) Standard Practice E 185, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels*. In the design of a plant-specific RV surveillance program, a licensee may use the edition of ASTM Standard Practice E 185, which was current on the issue date of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) to which the RV was purchased, or later editions through the 1982 edition.

As discussed in Section 3.3.1 of the PUSAR, Unit 1 has a plant-specific surveillance program which consists of three capsules. All three capsules have been in the RV since plant startup. In the PUSAR, TVA indicated that the uprate has no effect on the existing surveillance schedule. The removal of the first set of specimens from the RV is currently scheduled at the end of the first cycle after restart (U1C7 Refueling Outage), which most closely represents 8 effective full-power years (EFPYs) of operation. In accordance with the intent of ASTM E 185-82 and in compliance with Appendix H of 10 CFR Part 50, the remaining specimens would be withdrawn every 6 EFPYs after withdrawal of the first set of samples (i.e., at 14 EFPYs and 20 EFPYs, respectively). The NRC staff verified that TVA has adequately implemented a plant-specific RV surveillance program which meets the intent of ASTM E 185-82 and, therefore, is in accordance with Appendix H of 10 CFR Part 50.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed uprate on the RV surveillance withdrawal schedule and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the schedule. The NRC staff further concludes that the RV capsule withdrawal schedule is appropriate to ensure that the RV material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will provide the licensee with information to ensure continued compliance with draft GDC-9, 33, and 34 in this respect following implementation of the proposed uprate. Therefore, the NRC staff finds the RV material surveillance program acceptable with respect to the proposed uprate.

2.1.2 Pressure-Temperature Limits and Upper-Shelf Energy

Pressure-Temperature (P-T) limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences (AOOs) and hydrostatic tests. The NRC staff's review of P-T limits covered the P-T limits methodology and the calculations for the number of EFPYs specified for the proposed uprate, considering neutron embrittlement effects and using linear elastic fracture mechanics.

Regulatory Evaluation

Title 10 to the CFR Part 50, Appendix G, provides fracture toughness requirements for ferritic materials (low alloy steel or carbon steel) materials in the RCPB, including requirements on the upper shelf energy (USE) values used for assessing the safety margins of the RV materials against ductile tearing and requirements for calculating P-T limits for the plant.

The NRC's acceptance criteria for USE and P-T limits evaluations are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (2) draft GDC-33, insofar as it requires that the RCPB be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant; (3) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; (4) draft GDC-35 insofar as it requires that service temperatures for RCPB components constructed of ferritic materials ensure the structural integrity of such components when subjected to potential loadings; (5) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB and (6) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

Upper Shelf Energy Value Calculations

Title 10 to the CFR Part 50, Appendix G provides criteria for maintaining acceptable levels of USE for the RV beltline materials of operating reactors throughout the licensed life of the facility. Appendix G requires RV beltline materials to have a minimum USE value of 75 ft-lb in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb throughout the life of the facility, unless it can be demonstrated through analyses that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by

Appendix G of Section XI to the ASME Code. Appendix G also mandates that the methods used to calculate USE values must account for the effects of neutron irradiation on the USE values for the materials and must incorporate any relevant RV surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H RV materials surveillance program.

By letter dated April 30, 1993, the BWR Owners' Group (BWROG) submitted a topical report (TR) to document that BWR RVs would meet the margins of safety against fracture equivalent to those required by Appendix G to Section XI of the ASME Code for Charpy USE values less than 50 ft-lb. In a letter dated December 8, 1993, the NRC staff concluded that the TR demonstrates that the materials evaluated have the margins of safety against fracture equivalent to Appendix G to Section XI of the ASME Code, in accordance with Appendix G of 10 CFR Part 50. In this report, the BWROG derived through statistical analysis the initial USE values for materials that originally did not have documented Charpy USE values. Using these statistically-derived Charpy USE values, the BWROG predicted the end-of-life USE values in accordance with Position 1.2 in Regulatory Guide (RG) 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials*. According to RG 1.99, the decrease in USE is dependent upon the amount of copper in the material and the neutron fluence predicted for the material. The BWROG analysis determined that the minimum allowable Charpy USE in the transverse direction for base metal and along the weld for weld metal was 35 ft-lb.

GE performed an update to the USE equivalent margins analysis, which is documented in Electric Power Research Institute (EPRI) document TR-113596, *BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines*, BWRVIP-74, September 1999.

EPRI TR-113596 provides a bounding Charpy USE for BWR plants for 54 EFPYs. The analysis in EPRI TR-113596 determined the reduction in the unirradiated Charpy USE resulting from neutron radiation using the methodology in Position 1.2 in RG 1.99. Using this methodology and using a correction factor of 65 percent for conversion of the longitudinal properties to transverse properties, the lowest Charpy USE at 54 EFPYs for all BWR/3-6 plates is projected to be 45 ft-lb. The correction factor for specimen orientation in plates is based on SRP Branch Technical Position MTEB 5-2. Using the RG methodology the lowest Charpy USE at 54 EFPYs for shielded metal arc welds is projected to be 51.1 ft-lb. The value for the BWR/3-6 plates is greater than 35 ft-lb minimum allowable. This will meet the margins of safety against fracture equivalent to those required by 10 CFR Part 50, Appendix G. The value for the shielded metal arc weld is greater than the 50 ft-lb criteria in 10 CFR Part 50, Appendix G.

As stated in the license renewal application (LRA) for Unit 1, the fluence was calculated for the RV for the extended 60-year (54 EFPYs) licensed operating period, using the methodology of NEDC-32983P, *General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation*, which was approved by the NRC staff in an SE dated September 14, 2001. The licensee used one bounding fluence calculation for Unit 1.

Based on a fluence of 1.35×10^{18} n/cm^2 (E > 1.0 MeV), the licensee calculated the percent drop in USE for the limiting plate and weld. The percent drops in USE for the limiting plate and weld were 15.5 percent and 26.5 percent, respectively. These values are less than the allowable percent drop for the NRC staff-approved EPRI TR-113596 of 23.5 percent and 39 percent, respectively. The NRC staff independently confirmed the percent drop values for the Unit 1 limiting plate and weld. Therefore, the NRC staff concludes that the licensee has demonstrated that the Unit 1 RV complies with the requirements of 10 CFR Part 50, Appendix G through the end of its 60-year operating license.

Pressure-Temperature Limit Calculations

Section IV.A.2 of 10 CFR Part 50, Appendix G requires that the P-T limits for operating reactors be at least as conservative as those that would be generated if the methods of calculation in the ASME Code, Section XI, Appendix G were used to calculate the P-T limits. The rule also requires that the P-T limit calculations account for the effects of neutron irradiation on the RV beltline materials and to incorporate any relevant RV surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H RV materials surveillance program.

Section 3.3.1 of the PUSAR indicates that the P-T limit curves contained in the TSs would be revised considering the increases in shifts affecting the beltline portion of the curves.

By letter dated December 6, 2004, TVA submitted proposed changes related to the P-T limits in the Unit 1 TSs. The proposed amendment revised the Unit 1 RV P-T curves to reflect the results of an analysis which calculates the Unit 1 curves for 12 and 16 EFPYs of reactor operation. The Unit 1 P-T limit curves were approved by the NRC staff in an SE dated July 26, 2006. Therefore, the NRC staff finds the submitted P-T limit curves acceptable for use through 16 EFPYs.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed uprate on the USE values for the RV beltline materials and P-T limits for the plant. The NRC staff concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the USE values for the Unit 1 RV beltline materials and the P-T limits for the plant. The NRC staff concludes that the Unit 1 beltline materials will continue to have acceptable USE, as mandated by 10 CFR Part 50, Appendix G, through the expiration of the current facility operating license. As documented in the NRC staff's July 26, 2006, SE, the licensee has demonstrated the validity of the Unit 1 P-T limits for operation under uprated conditions through 16 EFPYs. Based on this, the NRC staff concludes that the Unit 1 RV will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with draft GDC-9, 33, 34, and 35 in this respect following implementation of the proposed uprate. Therefore, the NRC staff finds the proposed P-T limits acceptable with respect to the proposed uprate.

2.1.3 Reactor Internal and Core Support Materials

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other

SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement within both the fuel cladding and the reactor coolant system (RCS).

Regulatory Evaluation

The NRC staff's review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internal and core support materials are based on draft GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Specific review criteria are contained in SRP Section 4.5.2, BWRVIP-26, and Matrix 1 of RS-001.

Technical Evaluation

Reactor internals and core support materials are subject to the following degradation mechanisms:

- Cumulative fatigue damage
- Crack initiation and growth due to flow induced vibration
- Crack initiation and growth due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC) and irradiation assisted stress corrosion cracking (IASCC)
- Loss of fracture toughness due to thermal aging and neutron embrittlement

Cumulative fatigue damage is discussed in Section 2.2.3 and crack initiation and growth due to flow induced vibration are discussed in Sections 2.2.3 and 2.2.6 of this SE. Crack initiation and growth due to SCC and loss of fracture toughness due to thermal aging and neutron embrittlement are managed through the inservice inspection (ISI) program that conforms to the requirements of 10 CFR 50.55a and the BWR Vessel Internals Project (BWRVIP). The BWRVIP supplements the ISI program required by 10 CFR 50.55a. This program is reviewed and approved by the NRC.

The licensee indicated in their submittal that Unit 1 belongs to the BWRVIP organization and implementation of the procedurally-controlled program is consistent with the BWRVIP-issued documents. The inspection strategies recommended by the BWRVIP consider the effects of fluence on the applicable components and are based on component configuration and field experience. Reactor water chemistry conditions are maintained consistent with EPRI, BWRVIP, and established industry guidelines, except where technical justifications in accordance with the BWRVIP-94 report, *Program Implementation Guide*, have been documented. The licensee concluded that the current inspection program for the reactor internal components is adequate to manage any potential effects of uprated conditions because the increase in neutron fluence resulting from uprated conditions does not significantly increase the potential for degradation.

Note 1 in Matrix 1 of Section 2.1 of RS-001 indicates that guidance on the neutron irradiation-related threshold for inspection for IASCC in BWRs is provided in the BWRVIP-26 report, *BWR Top Guide Inspection and Flaw Evaluation Guidelines*. The *Final License*

Renewal SER for BWRVIP-26, dated December 7, 2000, states that the threshold fluence level for IASCC is 5×10^{20} n/cm^2 (E > 1 MeV).

Since uprated conditions do not significantly increase the potential for degradation, the NRC staff concludes that the current inspection program is acceptable for all RV internal components except for those that will exceed the threshold fluence level for IASCC.

The licensee stated in a December 19, 2005, letter that the components that will exceed the threshold fluence level for IASCC are the top guide, core shroud, core plate, and incore instrumentation dry tubes and guide tubes. The licensee based this on calculations performed in accordance with RG 1.190 to support the BFN, Units 1, 2, and 3 LRA. These components will be inspected and managed by the guidance in the BWRVIP and the Chemistry Control Program.

The NRC staff concludes that the BWRVIP and Chemistry Control Program are reasonable to manage the potential for IASCC of the top guide, core shroud, core plate, and incore instrumentation dry tubes and guide tubes.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed uprate on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. The NRC staff further concludes that the licensee has demonstrated that the reactor internal and core support materials will continue to be acceptable and will continue to meet the provisions of draft GDC-1 and 10 CFR 50.55a with respect to material specifications, welding controls, and inspection following implementation of the proposed uprate. Therefore, the NRC staff finds the evaluation of reactor internal and core support materials acceptable with respect to the proposed uprate.

2.1.4 Reactor Coolant Pressure Boundary Materials

The RCPB defines the boundary of systems and components containing the high-pressure fluids produced in the reactor. The NRC staff's review of RCPB materials covered their specifications, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs.

Regulatory Evaluation

The NRC's acceptance criteria for RCPB materials are based on (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-40 and 42, insofar as they require that protection be provided for engineered safety features (ESFs) against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of an LOCA; (3) draft GDC-9 and 33, insofar as they require that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (4) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; (5) draft GDC-35 insofar as it requires that service temperatures for RCPB components constructed of ferritic materials ensure the structural integrity of such components when subjected to potential loadings; and (6) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB. Specific review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for primary water stress-corrosion cracking of dissimilar metal welds and associated inspection programs is contained in Generic Letter (GL) 97-01, Information Notice 00-17, Bulletin (BL) 01-01, BL 02-01, and BL 02-02. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute (NEI), dated May 19, 2000.

Technical Evaluation

In the submittal, TVA stated that the reactor recirculation system (RRS) was generically evaluated in accordance with the process described in ELTR1 and ELTR2, and that an alternative piping evaluation process was used for the RCPB piping inside the primary containment.

The RCPB piping at Unit 1 that was evaluated for uprated conditions included the following systems: RRS, MS, reactor core isolation cooling (RCIC), high-pressure coolant injection (HPCI), FW, reactor water cleanup (RWCU), core spray (CS), standby liquid control SLC, RHR, RV head vent, RV bottom drain, MSRV discharge line, CRD hydraulic, and primary chemistry sampling. The licensee’s evaluation determined that the proposed uprate will not significantly affect the RCPB piping. The NRC staff finds the licensee’s conclusion acceptable because the above evaluation was performed in accordance with the processes identified in ELTR1 and ELTR2, which the NRC staff has previously reviewed and approved.

However, in its review of Section 3.5.1, Reactor Coolant Pressure Boundary Piping, of GE LTR NEDC-33006P, Revision 1, the NRC staff identified an Action Item that stated that power uprate applicants must identify all other than Category ‘A’ materials, as defined in NUREG-0313, Revision 2, that exist in its RCPB piping, and discuss the adequacy of the augmented inspection programs in light of the power uprate on a plant-specific basis.

In a letter dated June 6, 2005, TVA stated that consistent with the discussion in Section 3.6.1 of ELTR2, TVA has taken actions to identify, monitor, and mitigate IGSCC and will implement actions to monitor and mitigate flow-accelerated corrosion (FAC) in the Unit 1 RCPB prior to startup. The NRC staff’s evaluation of TVA’s FAC program is located in Section 2.1.6 of this SE. TVA further explained that for IGSCC to occur, three conditions must exist, namely the existence of a susceptible material, the presence of residual stress in the weld, and the presence of an aggressive environment. Operation at uprated conditions will result in somewhat higher pressure, temperature, and flow for some systems comprising portions of the RCPB, but these changes do not influence the causal factors required for IGSCC to occur. Operation at a higher power level will result in a slightly higher oxygen generation rate due to radiolysis of water; however, coolant chemistry will continue to be strictly controlled and maintained within specified limits. Therefore, operation at uprated conditions is expected to have a negligible impact on the occurrence of IGSCC.

The entire Unit 1 RRS piping has been replaced with corrosion-resistant material. This includes the pump suction and discharge piping, the ring header, the riser piping, and the inlet and outlet safe ends. The replacement piping and safe end material is Type 316 NG stainless material, which is resistant to IGSCC. The replacement piping utilized an improved design which eliminated several piping welds. The safe ends were replaced with an improved crevice-free design. As a result of these efforts, all the Unit 1 RRS welds are Category ‘A’ welds in accordance with NUREG-0313, Rev. 2 classifications. The use of IGSCC-resistant replacement materials and improved designs to reduce welds and crevices mitigate the possibility of future IGSCC.

The following piping was also replaced:

- The CS and RHR system piping inside the containment. The replacement piping is Type 316 NG stainless for the RHR system and ASME SA-333 Gr 6 high toughness grade carbon steel for the CS system, both of which are less susceptible to IGSCC.

- The RWCU system piping operating above 200 degrees F was replaced both inside and outside containment with Type 316 NG stainless steel, which is resistant to IGSCC.

- The jet pump instrumentation nozzle safe ends and seal assemblies were replaced with an improved design, fabricated from IGSCC-resistant Type 316 NG materials.

Additionally, TVA used EPRI welding techniques (such as machine welding where practical, reduced energy input, etc.) and will implement a Mechanical Stress Improvement Process to the RCPB welds to further reduce flaw propagation. The planned installation of a hydrogen water chemistry system will further reduce flaw initiation on IGSCC-susceptible stainless steel materials.

The NRC staff finds that TVA has established comprehensive plans to mitigate IGSCC. These plans include replacement of piping with IGSCC-resistant material, application of weld stress improvement measures, and implementation of hydrogen water chemistry at Unit 1.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed uprate on the susceptibility of RCPB materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in system operating temperature on the integrity of RCPB materials. The NRC staff further concludes that the licensee has demonstrated that the RCPB materials at Unit 1 will continue to be acceptable following implementation of the uprate and will continue to meet the provisions of draft GDC-1, draft GDC-9, draft GDC-33, draft GDC-34, draft GDC-35, draft GDC-40, draft GDC-42, 10 CFR Part 50, Appendix G, and 10 CFR 50.55a. Therefore, the NRC staff finds RCPB materials acceptable for operation at uprated conditions.

2.1.5 Protective Coating Systems (Paints) - Organic Materials

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities.

Regulatory Evaluation

The NRC staff's review covered paints used inside the containment for their suitability and stability during a design-basis LOCA (DBLOCA), considering radiation and chemical effects. The NRC's acceptance criteria for paints are based on (1) 10 CFR Part 50, Appendix B, which states quality assurance requirements for the design, fabrication, and construction of safety-related SSCs; and (2) RG 1.54, Revision 1, for guidance on application and performance monitoring of coatings in nuclear power plants. Specific review criteria are contained in SRP Section 6.1.2.

Technical Evaluation

The BFN Service Level 1 coatings are subject to the guidance of RG 1.54 and American National Standards Institute Standard N101.4, *Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities*, dated November 1972. The licensee stated that the qualification testing for Service Level 1 coatings used for new applications or repair activities inside containment meets the guidance of the standards listed above.

The licensee noted in their submittals that there have been changes in the pump flow rate, suction strainer approach velocity, and peak suppression pool temperature. These changes have been evaluated by the licensee using NRC approved techniques. Section 2.6.5 addresses the adequacy of suction strainer design and evaluation of NPSH required for ECCS pump operation. Considering the changes in the process variables above, the licensee's evaluation concluded that the amount of coatings transported to the suction strainer surface will be acceptable. Based on its reviews of the licensee's analyses, the NRC staff finds that protective coating debris will not hinder NPSH of the ECCS pumps.

The PUSAR did not address the potential impact of uprate on the coating system's ability to remain adhered to the substrate in the event of a DBLOCA. In a letter dated February 23, 2005, TVA stated that previous testing bounded the peak accident conditions for all Service Level 1 coatings inside containment with one exception. The coatings were tested to a gamma dose of 1.5×10^9 rads, which is greater than the 1×10^9 rads accumulated dose for the DBLOCA at 120-percent power. The peak pressure for the qualification testing was 70 psig, which exceeds the peak value calculated for a DBLOCA of 48.5 psig.

The temperature profile for the qualification testing had a peak value of 340 degrees F, which also exceeds the maximum calculated values for a DBLOCA (295.2 degrees F) and an MS line (MSL) break (MSLB) (336 degrees F). The licensee indicated that the zone of influence associated with a pipe break would not be impacted by the increased temperature and pressure of the RCS, as the previously analyzed zone of influence of 10 pipe diameters bounded the RCS temperature and pressure under uprated conditions.

The licensee, in its letter of February 23, 2005, indicated that one specific coating configuration had not been previously tested. In a letter dated December 19, 2005, the licensee indicated that the coating system was a feather edge overlap of Ameron 400NT over an existing coating. This combination of coatings had not previously been used in the Unit 1 containment. The results of qualification testing indicated that this coating system was not qualified for use at BFN. As a result of the qualification testing the licensee indicated that this coating system will not be used in the Unit 1 containment.

All unqualified coating systems are assumed to fail in a DBLOCA and be available for ECCS suction strainer blockage. Changes in environmental conditions caused by uprated conditions do not change assumed quantity of unqualified coating debris created in a DBLOCA. The quantity of unqualified coatings is tracked by the licensee to ensure that the total amount is less than that assumed in the ECCS strainer calculations.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects on protective coating systems and concludes that the licensee has appropriately addressed the impact of changes in conditions following a DBLOCA and their effects on the protective coatings. The NRC staff further concludes that the licensee has demonstrated that the protective coatings will continue to be acceptable and will continue to be bounded by qualification test conditions. Therefore, the NRC staff finds the protective coatings systems acceptable for operation at uprated conditions.

2.1.6 Flow-Accelerated Corrosion

FAC is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. Components made from stainless steel are immune to FAC and it is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur.

Regulatory Evaluation

The NRC staff's review of the effects of FAC and the adequacy of the licensee's FAC program focus on the licensee's ability to predict the rate of loss so that repair or replacement of damaged components could be made before they reach critical thickness. The licensee's FAC program is based on NUREG-1344, GL 89-08, and the guidelines in EPRI Report NSAC-202L-R2. It consists of predicting loss of material using the CHECWORKS computer code, and visual inspection and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

The licensee stated that the FAC program activities predict, detect, and monitor wall thinning in piping and components due to FAC. The FAC program uses selective component inspections as the basis for qualifying un-inspected components for further service. The licensee stated that a CHECWORKS FAC model has been developed for BFN to predict the FAC wear rate and the remaining service life for each piping component and that the CHECWORKS model is updated after each refueling outage. The FAC models are also used to identify FAC examination locations for the outage examination list.

The licensee indicated that all of the process variables affecting FAC (moisture content, temperature, oxygen, and flow velocity) would change as a result of the uprate. The licensee also indicated that all of these variables would remain within the bounds of the CHECWORKS FAC model parameters.

In a letter dated December 19, 2005, the licensee summarized the expected changes to the process variables listed above and the impact of each of the changes on the wear rate associated with FAC. In addition, the licensee provided a table of the most susceptible systems and the predicted increase in wear rate associated with each system. Because Unit 1 is in the process of a recovery effort following an extended shutdown, FAC data were not available. The licensee presented data from the Units 2 and 3 FAC program. Units 2 and 3 FAC is expected to be representative of Unit 1. The system that is predicted to experience the greatest increase in wear rate as a result of the uprate is the FWH drains from 3FWH to 4FWH. The increase in predicted wear associated with the heater drains is 19.4 percent and is due to an increase in temperature and flow rate.

In the December 19, 2005, letter, the licensee provided data for actual measured thickness versus CHECWORKS predicted thicknesses for 15 components in each Units 2 and 3. A table compared the actual measured thickness of each component with the thickness that was predicted by CHECWORKS during the previous outage. For nearly all components measured (27 out of 30) the measured wall thickness was greater than the thickness predicted by CHECWORKS. For the three components in which the wall thickness was less than the predicted wall thickness, the greatest variance was 4 percent. For nearly all components the predictive model conservatively calculated wear rates that were greater than the actual wear rates under current thermal operating conditions at 105-percent power. The accuracy of the predictive model for FAC is not expected to change as a result of the Unit 1 uprate since changes to all parameters influencing FAC will be accounted for in the CHECWORKS model.

The licensee evaluated the post-uprate operating conditions and the effect they will have on the current FAC program's ability to predict the remaining life of previously inspected components. The licensee has adjusted the scheduled inspections to account for changes in remaining component life based on uprated conditions.

Conclusions

The NRC staff has reviewed the licensee's evaluation of the effect of the proposed uprate on the FAC analysis for the plant and concludes that the licensee has adequately addressed changes in the plant operating conditions on the FAC analysis. The NRC staff further concludes that the licensee has demonstrated that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components. The NRC staff finds the licensee's FAC program acceptable for operation at uprated conditions.

2.1.7 Reactor Water Cleanup System

The RWCU system provides a means for maintaining reactor water quality by filtration and ion exchange and a path for removal of reactor coolant when necessary. Portions of the RWCU system comprise the RCPB.

Regulatory Evaluation

The NRC staff's review of the RWCU system included component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability; and the instrumentation and process controls for proper system operation and isolation. The review consisted of evaluating the adequacy of the plant's TSs in these areas under the uprated conditions. The NRC's acceptance criteria for the RWCU system are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (2) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (3) draft GDC-51, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement. Specific review criteria are contained in SRP Section 5.4.8.

Technical Evaluation

The licensee reviewed the RWCU system piping and components for operation at uprated conditions using the computer analysis model for RCPB systems. The system will operate at increased temperature and pressure under uprated conditions, however, the RWCU system flow will remain the same. The slightly increased temperature and pressure will not impede the RWCU system from performing its function of removing solid and dissolved impurities from the recirculated reactor coolant. The licensee stated that the FW flow will increase causing the ion concentration in the reactor water to increase. This change is considered insignificant with respect to the capability of the RWCU system and the purity of the reactor water will remain within the specified limits.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed power uprate on the RWCU system and concludes that the licensee has adequately addressed changes in impurity levels and pressure and their effects on the RWCU. The NRC staff further concludes that the licensee has demonstrated that the RWCU will continue to meet the provisions of draft GDC-9, draft GDC-51, and draft GDC-70. The NRC staff finds that uprated conditions will result in insignificant changes in the piping and components for RWCU, therefore, the RWCU system is acceptable for operation at uprated conditions.

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

SSCs important to safety could be impacted by the pipe-whip dynamic effects of a pipe rupture. The NRC staff conducted reviews of pipe rupture analyses to ensure that SSCs important to safety are adequately protected from the effects of pipe ruptures. The NRC staff's review covered (1) the implementation of criteria for defining pipe break and crack locations and configurations; (2) the implementation of criteria dealing with special features, such as augmented ISI programs or the use of special protective devices such as pipe-whip restraints; (3) pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects; and (4) the design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings. The NRC staff's review focused on the effects that the proposed power uprate may have on items (1) through (4) above. The NRC's acceptance criteria are based on draft GDC-40, insofar as it requires that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures. Specific review criteria are contained in SRP Section 3.6.2.

Technical Evaluation

The design basis for the original Unit 1 RCPB piping, components and supports systems include postulated breaks in all high-energy piping above 1 inch in diameter. For 105-percent power uprate, the maximum RV dome pressure increases from 1005 psig to 1035 psig (increase of 30 psi). There is no significant increase in temperature (less than 4 degrees F) and flow rate in the RCPB piping except the FW and MSLs where the flow rate increase is about 6 percent. The licensee determined that the increase in flow rate during normal operation at uprate conditions has no effect on the mass, energy releases and the break flow velocity, since they are determined by reactor pressure (which remains below 1250 psi design pressure), and the size of the pipe (which is unchanged). Therefore, the loads associated with the thrust at the break locations, jet impingement loadings at and away from the break locations, and asymmetric pressurization remain unchanged for the 105-percent OLTP conditions.

The licensee reviewed pipe stresses and fatigue usage factor calculations for the as-built configurations of Unit 1 piping systems at EPU conditions, which bound the 105-percent power conditions. The loads in the piping structural evaluation include seismic loads, thermal loads, SRV discharge loads, and LOCA loads including pool swell, condensation oscillation, and chugging loads. The seismic loads are not affected by the power uprate. As a result of review, the licensee determined that the uprate conditions are bounded by the DBLOCA loads based on the test conditions defining the pool swell, condensation oscillation, and chugging loads. The licensee also determined that the parameters used to define the SRV loads are not affected by the uprate and, therefore, the existing SRV loads for Unit 1 remain applicable at 105-percent power uprate conditions. No new postulated pipe break locations were identified by the licensee.

On the basis of its review, the NRC staff finds the licensee's analysis methodology associated with the break locations and the associated dynamic effects of SRV and LOCA loads to be consistent with SRP section 3.9.3 and the analysis results acceptable for operation at 105-percent OLTP.

Conclusion

The NRC staff has reviewed the licensee's evaluations related to determinations of rupture locations and associated dynamic effects and concludes that the licensee has adequately addressed the effects of the proposed uprate. The NRC staff further concludes that the licensee has demonstrated that ESFs will continue to meet the provisions of draft GDC-40 following implementation of the uprate. Therefore, the NRC staff finds the licensee's evaluation of rupture locations and dynamic effects associated with the postulated rupture of piping acceptable for the proposed power uprate.

2.2.2 Pressure-Retaining Components and Component Supports

Regulatory Evaluation

The NRC staff has reviewed the structural integrity of pressure-retaining components (and their supports) designed in accordance with ASME Code, Section III, Division 1, and draft GDC-1, 2, 9, 33, 40 and 42. The NRC staff's review focused on the effects of the proposed power uprate on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions. The NRC staff's review covered (1) the analyses of flow-induced vibration and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and cumulative fatigue usage factors (CUFs) against the code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of an LOCA; and (4) draft GDC-9 and 33, insofar as they require that the RCPB be designed and constructed so as to have an exceedingly low probability of RCPB gross rupture or significant leakage; and (5) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 5.2.1.1; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

Nuclear Steam Supply System Piping, Components, and Supports

The RCPB piping system consists of a number of safety-related piping subsystems that move fluid through the reactor and other safety systems. The licensee evaluated the effects of the power uprate condition, including higher flow rate, temperature, pressure, fluid transients and vibration effects on the RCPB and balance-of-plant (BOP) piping systems and components. The components evaluated included equipment nozzles, anchors, guides, penetrations, pumps, valves, flange connections, and pipe supports (including snubbers, hangers, and struts). The licensee indicated that the original code of record as referenced in the original and existing design basis analyses was used in the evaluation. The NRC staff finds this acceptable.

The RCPB piping systems evaluated included the RRS, MS, MS drains, RCIC, HPCI, FW, RWCU, CS, SLC, RHR, RV head vent line, CRD piping, and SRV discharge line systems. The evaluation for pipe stresses used the United States of America Standards (USAS) B31.1, *Power Piping* (1967 Edition), which is the Unit 1 code of record for the design basis of piping.

The licensee indicated that the evaluation follows the process and methodology defined in Appendix K of ELTR1 and in Section 4.8 of Supplement 1 of ELTR2.

In general, the licensee compared the increase in pressure, temperature and flow rate due to the power uprate against the same parameters used as input to the original design-basis analyses. The comparison resulted in bounding percentage increases in stresses for affected limiting piping systems. The bounding percentage increases are compared to the design margin between calculated stresses and the Code allowable limits. The bounding percentage increases were also applied to the original calculated stresses for the piping to determine the stresses at the proposed power uprate condition. The NRC staff finds the methodology to be acceptable considering the conservatism in the application of the scaling factors for the power uprate stress-to-loading combinations that include individual loads (i.e., dead weight and seismic) that are not affected by the power uprate.

The NRC staff noted that Unit 1 is currently performing restart modifications and final stress results which reflect the final as-built configuration are not available. In a letter dated July 26, 2006, the licensee provided maximum calculated stresses for the limiting FW and MS piping systems at Units 2 and 3 for the 105-percent power condition. The maximum stresses shown in the tables are less than the code allowable limits for both the FW and MS piping systems. Based on the similarity of the MS and FW piping between the units the NRC staff determined the stresses in MS and FW piping will be within the code allowable limits for the 105-percent power uprate.

At 105-percent power uprate conditions, the flow, pressure, temperature, and mechanical loading for the RCPB piping systems remain either the same or change insignificantly (i.e., within the safety margin). The licensee evaluated the MS and FW lines and associated branch piping systems in accordance with the requirements of USAS-B31.1 (1967 Edition) for the effects of the uprate on piping, piping supports including the associated building structure, piping interfaces with the reactor pressure vessel (RPV) nozzles, penetrations, flanges and valves. The increase in MS flow results in increased forces from the turbine stop valve closure transient. The turbine stop valve closure loads bound the MSIV closure loads because the MSIV closure time is significantly longer than the stop valve closure time.

Seismic inertia loads and seismic building displacement loads are not affected by the 105-percent power uprate. There is no effect on the analyses for these load conditions. The licensee performed a bounding piping analysis including the effects of the uprate conditions. The licensee evaluated piping supports such as snubbers, hangers, struts, anchorages, equipment nozzles, guides, and penetrations by evaluating the piping interface loads due to the increases in pressure, temperature, and flow for affected limiting piping systems. The evaluation shows that there is adequate design margin between the original design stresses and code limits for the supports to accommodate the load increase due to the proposed 105-power uprate. The NRC staff finds the evaluation methodology to be acceptable considering the conservatism in the application of the scaling factors for the power uprate stress to loading combinations that include individual loads (i.e., dead weight and seismic) that are not affected by the power uprate.

Piping systems other than FW and MSLs connecting to the RCPB do not experience an increase in flow rate at uprate conditions. The normal operating pressure and temperature of the reactor are slightly changed for the power uprate. The licensee evaluated these systems by reviewing the original design basis analysis of record. The review shows that there is adequate design margin between the original design stress and the Code limits. Therefore, these piping systems continue to comply with the USAS B31.1 Code and are acceptable by the NRC staff to operate following the 105-percent power uprate.

In a letter dated December 19, 2005, the licensee indicated that the piping vibration monitoring program for Unit 1 addresses flow-induced vibration (FIV) of the critical piping systems that will experience increased flow during uprated conditions. The piping steady state vibration program for EPU operation follows the guidelines of ASME Operations and Maintenance Code, Part 3, Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Piping Systems. The program will assess the flow-induced steady state vibration levels of selected piping systems that will experience increased flow during 105-percent power operating conditions. The program will include branch lines and cantilevered small bore lines which industry experience has shown are vulnerable to high-cycle fatigue failures.

On the basis of its evaluation, the licensee determined that, for all RCPB piping systems, the original piping design has sufficient design margin to accommodate the changes due to the proposed power uprate. The NRC staff reviewed relevant portions of the evaluation provided by the licensee and finds the licensee’s conclusion to be acceptable for a 105-percent OLTP power uprate.

<u>Balance-of-Plant Piping, Components, and Supports</u>

The licensee’s evaluations of the stresses for BOP piping and related components, connections and supports are similar to the evaluation of the RCPB piping and supports. The BOP systems evaluated by the licensee include lines which are affected by the power uprate, but not evaluated in Section 3.5 of the PUSAR. The existing design analyses of the affected BOP piping systems were reviewed against the uprated power conditions. As a result of its evaluation, the licensee concluded that there are sufficient margins in the original design analyses to accommodate the changes due to the proposed power uprate and, therefore, all piping meets the requirements of USAS B31.1, 1967 Edition, which is the code of record.

At Unit 1, the reactor building BOP piping evaluation effort, in addition to the power uprate effects, included changes due to NRC BL 79-14 walkdown as-built data, seismic design criteria and spectra changes, and the piping and component replacement changes. Other items including penetration anchors, RPV nozzles, flanges, and commodity clearances were also evaluated for effects by the analysis results. Certain torus piping calculations had small changes which did not require the full piping analysis to be redone, however, the previous analysis results were scaled up by using the multiplying factors based on the increase in temperatures. The resulting stresses were verified within the code allowable limit. The torus-attached piping stress calculations were revised to document power uprate changes and evaluation results.

In a July 26, 2006, letter, the licensee indicated that Unit 1 is currently performing restart modifications and the final stress results, which reflect the final as-built configuration, are not available for most of the piping systems. In the July letter, the licensee provided the calculated maximum stresses and allowable stress limits for the critical BOP piping systems at Units 2 and 3 for the 120-percent power uprate. The provided stress-to-allowable ratios indicate that there will be sufficient margins between actual stresses and Code allowable limits in the original design to accommodate the slight increases in temperature (less than 4 degrees F), pressure (less than 30 psi) and flow rate (less than 6 percent) resulting from the 105-percent power uprate condition at Unit 1 which is similar to Units 2 and 3 in design.

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. The piping systems attached to the torus shell are designed for a temperature limit of 177 degrees F, which remains the same for the power uprate. The seismic loadings are not affected by the power uprate. The licensee concluded that all piping is below the code-allowable limits. The NRC staff reviewed the licensee's submittals and finds that the licensee's evaluations in conjunction with the operating experience of Units 2 and 3 at the 105-percent OLTP power, provide reasonable assurance for a safe operation of Unit 1 at 105-percent OLTP power.

The licensee evaluated the FIV levels of the safety-related MS and FW piping systems that are projected to increase in proportion to the increase in the fluid density and the square of the fluid velocity following the proposed power uprate. The NRC staff's evaluation of the licensee's FIV program is provided in Section 2.2.6.

Regarding the assessment of the MS flow restrictor, the licensee stated that there is no impact on the structural integrity of the restrictor as a result of the proposed power uprate. In Section 3.1 of the September 22, 2006, submittal, the licensee indicated that a peak RV dome pressure of 1301 psig results from the overpressure protection event analysis for the proposed uprate conditions, but this value remains below the ASME Code limit of 1375 psig (110-percent of design pressure). Also, the restrictor was designed for a maximum differential pressure due to the choke flow condition, which is bounding for the uprated power condition. Therefore, the MSL flow restrictor will maintain its structural integrity following the power uprate. The licensee evaluated the MSIVs by referring to the GE generic evaluation in Section 4.7 of ELTR-2, which is applicable to the proposed uprate. The licensee determined that the existing design pressure and temperature for the MSIVs are bounding for the proposed power uprate and that the ability of the MSIVs to perform their isolation function is not affected following the power uprate condition.

Based on the above review, the NRC staff finds that the design of BOP piping, components and their supports is adequate to ensure that the BOP system will maintain its structural and pressure boundary integrity for the 105-percent OLTP power operation at Unit 1.

Reactor Vessel and Supports

The licensee evaluated potential effects of the Unit 1 power uprate on the RV and internal components in accordance with its current design basis. The loads considered in the evaluation include reactor internal pressure difference, LOCA, flow loads, acoustic loads, thermal loads, seismic, and dead weight. The licensee indicated that the load combinations for normal, upset, emergency, and faulted conditions were considered consistent with the current design basis analysis. In its evaluation, the licensee compared the proposed power uprate conditions for pressure, temperature and flow against those used in the design basis. For cases where the power uprate conditions are bounded by the design basis analyses, no further evaluation was performed. If the power uprate conditions were not bounded by the design basis, new stresses were determined by scaling up the existing design basis stresses proportional to the proposed power uprate conditions. The resulting stresses are compared against the applicable allowable values (AVs), in accordance with the design basis. The NRC staff finds the methodology used by the licensee to be consistent with the NRC-approved methodology in Appendix I of ELTR1, and is therefore acceptable.

The stresses and CUFs for the RV components were evaluated by the licensee in accordance with the ASME Code, Section III, 1965 Edition with addenda up to and including summer 1965, which is the Code of Record at Unit 1. For evaluation of the power uprate, a scaling factor was developed based on the increase in pressure, temperature, and flow rate to recalculate the stresses in the RV components in accordance with the method described in Appendix I of ELTR1, which has been previously approved by the NRC staff in a letter dated February 8, 1996. The evaluation methodology is considered conservative in the application of the scaling factors for the power uprate stress to loading combinations that include individual loads (i.e., dead weight and seismic) that are not affected by the power uprate. Due to the conservative method used to recalculate these resultant stress and fatigue usage factors, the actual design margin with respect to the ASME Code limits for the proposed power uprate limiting RV pressure boundary stress condition is conservatively underestimated.

The licensee indicated that the maximum primary plus secondary stresses for critical components such as FW nozzle, recirculation outlet nozzle, main closure stud, and support skirt of the RV level were calculated by using the power uprate scaling factor. For these limiting components, the calculated CUFs and the maximum stresses provided in Table 3-4 of the PUSAR for the uprated power conditions are within the code allowable limits and therefore acceptable.

The RV components that are not listed in Table 3-4 of the PUSAR have maximum stresses and CUFs that are either not affected by the power uprate or are already bounded by those listed in the table. Additionally, in accordance with the proposed power uprate methodology in Appendix I of ELTR1 for addressing plant normal and upset operational conditions, an evaluation of fatigue is necessary only for those RV components having a CUF greater than

0.5. This methodology has been previously approved by the NRC staff and therefore is acceptable.

The primary plus secondary stresses and the CUF results presented in Table 3-3 of the PUSAR demonstrate that the RV pressure boundary and pressure boundary penetrations, including the FW nozzle penetration, are less than the ASME Code, Section III stress and fatigue usage factor AVs. The maximum stresses for critical components of the RV internals were summarized in Table 3-8 of the PUSAR for the currently licensed power level and the proposed power uprate conditions. These calculated stresses are also less than the allowable Code limits. The NRC staff's evaluation of the structural integrity of the RV internals is discussed in Section 2.2.3.

Based on its review of the licensee's evaluation of the RV and internals, the NRC staff finds that the maximum stresses and fatigue usage factors are within the Code-allowable limits. The NRC staff also finds that the RV and internals will continue to maintain their structural integrity for the power uprate condition.

Control Rod Drive Mechanism

In Section 2.5.3 of the PUSAR, the licensee indicated that the pressure boundary components of the CRD system have been designed in accordance with the Code of Record, the ASME Code, Section III, 1974 Edition up to and including the Winter 1975 Addenda, and that the original design basis analysis for the CRD system remains unchanged by the proposed power uprate conditions. The components of the CRD system, which form part of the primary pressure boundary, have been designed for a bottom head pressure of 1250 psig, which is higher than the analytical limit of 1070 psig for the reactor bottom head pressure. For the power uprate condition, the vessel bottom head temperature increases by approximately 4 degrees F to 532 degrees F, which is bounded by the CRD Mechanism (CRDM) design temperature. There is no change on the seismic loading and fuel lift loads for the power uprate. Therefore, the primary plus secondary stress and the CUFs for the CRD housing remain the same as those calculated for the currently licensed power level, which are less than the ASME Code, Section III AVs.

In addition, 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 caused by a postulated abnormal operating condition. The analysis of cyclic operation of the CRDM resulted in a maximum CUF for the limiting CRD main flange to be less than the code-allowable CUF limit of 1.0 for the power uprate.

On the basis of its review, the NRC staff finds that the CRDMs will continue to meet its design basis and performance requirements at the proposed power uprate conditions.

Recirculation Pumps and Supports

At the 105-percent power uprate conditions, the slight increase in flow (6 percent), pressure (3 percent), and temperature (less than 1 percent) for the RCPB piping systems is considered insignificant (i.e., within the safety margin). For the power uprate operation, the core flow rate remains unchanged. At current rated core flow, the recirculation pump flow will slightly increase by about 2 percent of rated pump flow to accommodate the higher flow resistance at uprated conditions. The licensee reviewed the stress and fatigue calculation in the analysis of record for the current design basis of the recirculation piping and pumps, and determined that there is sufficient margin to the Code-allowable limits to accommodate the 2-percent increase in pump flow rate. Consequently, the licensee determined that the uprated conditions are within the original design capability of the system equipment including the pump, valves, piping systems and supports. Based on its review, the NRC staff finds that the current design of the recirculation piping system (including pumps and supports) is adequate to operate at the 105-percent power uprate conditions.

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of pressure-retaining components and their supports. For the reasons set forth above, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed uprate on these components and their supports. Based on the above, the NRC staff further concludes that the licensee has demonstrated that pressure-retaining components and their supports will continue to meet the requirements of 10 CFR 50.55a, draft GDC-1, 2, 9, 33, 34, 40, and 42 following implementation of the proposed uprate. Therefore, the NRC staff finds the structural integrity of the pressure-retaining components and their supports acceptable for operation at power uprate conditions.

2.2.3 Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

RV internals consist of all the structural and mechanical elements inside the RV, including core support structures. The NRC staff reviewed the effects of the proposed uprate on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with LOCAs, and the identification of design transient occurrences. The NRC staff's review covered (1) the analyses of FIV for safety-related and non-safety-related reactor internal components and (2) the analytical methodologies, assumptions, ASME Code editions, and computer; programs used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of an LOCA; and (4) draft GDC-6, insofar as it requires that the reactor core be designed with appropriate margin to assure that acceptable fuel damage limits are not exceeded during any condition of normal operation, including the effects of AOOs. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5, and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

The licensee evaluated effects of the Unit 1 power uprate on the RV and internal components in accordance with its current design basis. The steam dryer is addressed in Section 2.2.6. The loads considered in the evaluation include reactor internal pressure difference, LOCA, flow loads, acoustic loads, thermal loads, seismic, and dead weight. The licensee indicated that the load combinations for normal, upset, emergency, and faulted conditions were considered consistent with the current design basis analysis. In its evaluation, the licensee compared the proposed power uprate conditions for pressure, temperature and flow against those used in the design basis. For cases where the power uprate conditions are bounded by the design basis analyses, no further evaluation is performed. If the power uprate conditions are not bounded by the design basis, new stresses are determined by scaling up the existing design basis stresses proportionate to the proposed power uprate conditions. The resulting stresses are compared against the applicable AVs, in accordance with the design basis. The NRC staff finds the methodology used by the licensee consistent with the NRC-approved methodology in Appendix I of ELTR1, and is therefore acceptable.

The stresses and CUFs for the RV components were evaluated by the licensee in accordance with the ASME Code, Section III, 1965 Edition with addenda up to and including Summer 1965, which is the Code of Record at Unit 1. The licensee indicated that the reactor internal components are not ASME Code components, except the CRD, as noted, however, the requirements of the code are used as guidelines in their design basis analysis. The licensee also indicated that the evaluations supporting the thermal power increase were performed consistent with the Unit 1 design basis. This is acceptable to the NRC staff.

The licensee provided the calculated maximum stresses and CUFs for the most limiting RV components in the PUSAR. The RV components that are not listed have maximum stresses and CUFs that are either not affected by the power uprate or already bounded by those listed in the table. The maximum calculated stresses are within the Code-allowable limits, and the CUFs are less than the Code limit of 1.0. The maximum stresses for critical components of the reactor internals are less than the allowable Code limits and, therefore, acceptable.

In its assessment of the potential for FIV on the reactor internals components, the licensee indicated that the steam separators and dryers in the upper elevations of the reactor are the components most affected by the increased steam flow due to the proposed power uprate. The effects of the power uprate on the FIV for other components in the reactor annulus and core regions are less significant because the proposed power uprate conditions do not require any increase in core flow, and very little increase in the drive flow.

For components other than the steam separators and dryers, the evaluation of FIV for the reactor internal components was performed based on the vibration data recorded in Unit 1 or similar plants and on the GE BWR operating experience. The licensee indicated that the evaluation was conservatively based on a reactor power of 3952 MWt and 105-percent of the rated flow. The vibration levels were calculated by extrapolating the recorded vibration data to power uprate conditions and compared to the plant allowable limits. The stresses at critical locations were calculated based on the extrapolated vibration peak response displacements and found to be within the GE allowable design criteria of 10 kilo-pounds per square inch (ksi) (where 1 ksi = 1000 pounds per square inch (psi)). Stress values less than 10 ksi for stainless steel are within the endurance limit under which sustained operation is allowed without incurring any cumulative fatigue usage. The vibration evaluation methodology, as described in Section 3.3.5 of the PUSAR, is conservative based upon the absolute sum combination of the various modes of vibration, including the absolute sum of the maximum vibration amplitude occurring in each mode. The licensee concluded that vibration levels of all safety-related reactor internal components are within the acceptance criteria. The NRC staff finds the licensee's specified stress limit of 10 ksi for the reactor internal components to be reasonably conservative in comparison to the ASME Code limit of 13.6 ksi for the peak vibration stress and is, therefore, acceptable.

In the PUSAR, the licensee indicated that the steam separators and dryer are not safety-related components; however, their failure may lead to an operational concern. In a July 26, 2006, letter, the licensee assessed the capability of the steam separators by extrapolation based on the OLTP data considering the turbulence and the periodical excitation due to swirling motion of the steam through the separator tubes. As a result, the separator vibration amplitudes were calculated to increase by 58-percent from OLTP for a reactor power of 3952 MWt and resulted in a maximum stress of 3,800 psi, which is less than the allowable limit of 10,000 psi. On the basis of information provided by the licensee, the NRC staff concludes that the licensee has reasonably demonstrated that the steam separators will meet their design basis requirements and maintain their structural integrity under uprated conditions.

The licensee's analysis for the structural integrity of the steam dryer assembly for the FIV loading at 105-percent OLTP is addressed separately in Section 2.2.6.

Based on its review of the licensee's evaluation of the RV and internals, the NRC staff finds that the maximum stresses and fatigue usage factors are within the Code-allowable limits. The NRC staff also finds that the RV and internals will continue to maintain their structural integrity for the 105-percent OLTP power operation.

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of reactor internals and core supports and concludes that the licensee has adequately addressed the effects of the proposed uprate on the reactor internals and core supports. The NRC staff further concludes that the licensee has demonstrated that the reactor internals and core supports will continue to meet the provisions of 10 CFR 50.55a, draft GDC-1, 2, 6, 40, and 42 following implementation of the proposed uprate. Therefore, the NRC staff finds the design of the reactor internal and core supports acceptable with respect to the proposed uprate.

2.2.4 Safety-Related Valves and Pumps

Regulatory Evaluation

The NRC's staff's review included certain safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME Code and within the scope of Section XI of the ASME Code and the ASME Operations and Maintenance Code, as applicable. The NRC staff's review focused on the effects of the proposed uprate on the required functional performance of the valves and pumps. The review also covered any impacts that the uprate may have on the licensee's motor-operated valve (MOV) programs related to GL 89-10, GL 96-05, and GL 95-07. The NRC staff also evaluated the licensee's consideration of lessons learned from the MOV program and the application of those lessons learned to other safety-related power-operated valves. The NRC's acceptance criteria are based on (1) draft GDC-1, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-38, 46, 47, 48, 59, 60, 61, 63, 64, and 65 insofar as they require that the ECCS, the containment heat removal system, the containment atomospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components; (3) draft GDC-57, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the inservice testing (IST) program requirements identified in that section. Specific review criteria are contained in SRP Sections 3.9.3 and 3.9.6; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

When GL 89-10 was issued in 1989, TVA did not implement the recommendations of the GL to verify the design-basis capability of safety-related MOVs at Unit 1 because it was in a long-term shutdown. In a submittal dated May 4, 2004, TVA reported that an MOV program had been developed at Unit 1 in response to GL 89-10 in preparation for plant restart. According to that submittal, the review and documentation of the design basis for the operation of each GL 89-10 MOV at Unit 1; scope of the GL 89-10 program; methods for determining and adjusting MOV switch settings; testing; surveillance; and maintenance were the same as within the GL 89-10 program at Units 2 and 3. TVA stated that the MOV switch settings at Unit 1 would be set prior to restart, but that some dynamic testing would be conducted during power ascension. TVA also indicated its commitment to implement the Joint Owners' Group Program on MOV periodic verification as part of its response to GL 96-05.

In a letter dated December 19, 2005, TVA described the ongoing implementation of the GL 89-10 program at Unit 1. The design and modification activities include (1) evaluations and changes necessary to support fulfillment of the GL 89-10 program recommendations; (2) evaluations and changes necessary to support operation of Unit 1 at EPU conditions; and (3) evaluations and modifications necessary to support closure of any other restart commitments potentially affecting GL 89-10 MOVs. TVA stated that 17 MOVs will be entirely replaced and 34 MOV actuators will be replaced. TVA established a goal to have all GL 89-10 MOVs equipped with SMARTSTEMS to facilitate diagnostic testing. The GL 89-10 MOVs at Unit 1 will be tested as part of the post-modification program before being declared operable.

In the September 22, 2006, request, TVA stated that based on test requirements and system configurations it would be necessary to perform differential pressure testing of some MOVs after restart. TVA committed to perform the GL 89-10 testing within 30 days following completion of the power ascension test program at Unit 1.

On November 28-30, 2006, the NRC staff conducted an inspection of the MOV program at Unit 1. The staff found the MOV program at Unit 1 to be well developed with reasonable design assumptions and consideration of industry operating experience. TVA was in the process of completing its GL 89-10 program during the inspection. For the long-term program in response to GL 96-05, TVA will implement the industry-wide Joint Owners Group Program on MOV Periodic Verification, which was accepted in an NRC staff evaluation dated September 25, 2006. The NRC staff's walkdown inspection found the MOVs that have been readied for operation to be in good condition. TVA will notify the NRC when the GL 89-10 program is complete.

With respect to GL 95-07 to address potential pressure locking and thermal binding of safety-related power-operated valves at Unit 1, the NRC staff has previously issued an SE concluding that TVA was implementing an acceptable response to GL 95-07. As part of its GL 95-07 program, in its submittal dated September 22, 2006, TVA indicated that one HPCI valve and two CS minimum flow valves at Unit 1 will have double disc valves installed prior to restart. TVA will also modify five safety-related power-operated gate valves by drilling a hole in the reactor side disc to preclude the potential for pressure locking.

As discussed in Section 3.7 of the PUSAR, the increase in steam flow under power uprate conditions will assist in the closure of the MSIVs at Unit 1. In its submittal dated December 19, 2005, TVA described the self-compensating feature of the hydraulic control process that will maintain closing time with little deviation despite the change in steam flow. In addition, the licensee established margin in the closure time criteria for MSIV testing to account for potential reduction in stroke time from the increased steam flow.

In Section 4.2 of the PUSAR, TVA discussed its evaluation to demonstrate that ECCS performance at Unit 1 will remain acceptable under uprated conditions. TVA determined that the safety-related pumps in the HPCI system were not impacted by uprated conditions. However, TVA has requested approval to rely on 3 psi containment overpressure to support NPSH for specific ECCS pumps. This discussion is contained in section 2.6.5 of this SE.

In its submittal dated December 19, 2005, TVA described the review of the IST Program for safety-related pumps and valves at Unit 1 for EPU operations. The Code of Record for Unit 1 is the 1995 Edition through 1996 Addenda of the ASME OM Code. The IST Program at Unit 1 assesses the operational readiness of pumps and valves within the scope of the ASME OM Code. The scope of the IST Program at Unit 1, and the testing frequencies, will not be affected by the power uprate. No changes in the IST Program at Unit 1 in support of the power uprate request are anticipated with the exception of specific implementing procedures.

As described in the licensee's submittal dated December 19, 2005, the air-operated valves (AOVs) and solenoid-operated valves (SOVs) used for containment isolation at Unit 1 were evaluated for potential effects from EPU conditions. For these valves, performance will be bounded by the design inputs, analytical scenarios, and methodologies of existing analyses. Existing design pressure and temperatures were determined to be adequate for these valves. As a result, the capability of AOVs and SOVs used for containment isolation at Unit 1 to perform their containment isolation function under power uprate conditions was confirmed.

In its submittal dated April 25, 2005, TVA described the modifications planned for Unit 1 in support of the EPU request. Many of the modifications are related to the changes in MS and

FW operating parameters. For example, condensate, condensate booster, and reactor FW modifications are being performed to upgrade the components to provide the higher flows for EPU operating conditions. Many of the GL 89-10 MOVs will be replaced with the remainder being refurbished. The NRC staff reviewed the status of the GL 89-10 MOV modifications at Unit 1 as part of the NRC inspection conducted on November 28-30, 2006 .

Conclusion

The NRC staff has reviewed the licensee's assessments related to the functional performance of safety-related valves and pumps and concludes that the licensee has adequately addressed the effects of the proposed power uprate on safety-related pumps and valves. The NRC staff further concludes that the licensee has adequately evaluated the effects of the proposed power uprate on its MOV programs related to GL 89-10, GL 96-05, and GL 95-07, and the lessons learned from those programs to other safety-related, power-operated valves. Based on this, the NRC staff concludes that the licensee has demonstrated that safety-related valves and pumps will continue to meet the requirements of draft GDC-1, 38, 46, 47, 48, 57, 59, 60, 61, 63, 64, and 65, and 10 CFR 50.55a(f) following implementation of the proposed power uprate. Therefore, the NRC staff finds the functional performance of safety-related valves and pumps acceptable with respect to the proposed power uprate.

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant releases of radioactive materials to the environment are also covered by this section.

Regulatory Evaluation

The NRC staff's review focused on the effects of the proposed power uprate on the qualification of the equipment to withstand seismic events and the dynamic effects associated pipe-whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake are not affected by a power uprate. The NRC's acceptance criteria are based on (1) draft GDC-1, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) 10 CFR Part 100, Appendix A, which sets forth the principal seismic and geologic considerations for the evaluation of the suitability of plant design bases established in consideration of the seismic and geologic characteristics of the plant site; (4) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of an LOCA; (5) draft GDC-9 and 33, insofar as they require that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (6) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; and (7) 10 CFR Part 50, Appendix B, which sets quality assurance requirements for safety-related equipment. Specific review criteria are contained in SRP Section 3.10.

Technical Evaluation

The licensee evaluated equipment qualification for the power uprate condition at Unit 1. The Unit 1 plant-specific dynamic loads such as SRV discharge and LOCA loads (including pool swell, condensation oscillation, and chugging loads) that were used in the equipment design will remain unchanged as discussed in Section 4.1.2 of the PUSAR. This is because these loads are based on the range of test conditions for the design-basis analysis at Unit 1, which are bounding for the 105-percent OLTP power operating condition.

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

- The seismic loads are unaffected by the power uprate;
- No new pipe break locations or pipe whip and jet impingement targets are postulated as a result of the uprated condition;
- Pipe whip and jet impingement loads do not increase for the power uprate; and
- SRV and LOCA dynamic loads used in the original design basis analyses are bounding for the power uprate.

Conclusion

The NRC staff has reviewed the licensee's evaluations of the effects of the proposed power uprate on the qualification of mechanical and electrical equipment and concludes that the licensee has (1) adequately addressed the effects of the proposed power uprate on this equipment and (2) demonstrated that the equipment will continue to meet the provisions of draft GDC-1, 2, 9, 33, 34, 40, and 42; 10 CFR Part 100, Appendix A; and 10 CFR Part 50, Appendix B, following implementation of the proposed power uprate. Therefore, the NRC staff finds the qualification of the mechanical and electrical equipment acceptable with respect to the proposed power uprate.

2.2.6 Additional Review Area - Potential Adverse Flow Effects

Plant operation at power uprate conditions can result in adverse flow effects on the MS, FW, and condensate systems and their components (including the steam dryer in BWR plants) from increased system flow and FIV. Some plant components, such as the steam dryer, do not perform a safety function, but must retain their structural integrity to avoid the generation of loose parts that might adversely impact the capability of other plant equipment to perform their safety functions.

Regulatory Evaluation

The NRC staff reviewed the licensee's consideration of potential adverse flow effects of the proposed 105-percent power uprate at Unit 1, including consideration of the design input parameters and the design-basis loads and load combinations for the steam dryer for normal operation, upset, emergency, and faulted conditions. The NRC staff's review covered the analytical methodologies, assumptions, and computer programs used in the evaluation of the steam dryer. The NRC staff's review included a comparison of the resulting stresses against applicable limits. The NRC staff also reviewed the licensee's evaluation of other reactor, MS, FW, and condensate system components for potential susceptibility to adverse flow effects from power uprate operation. The NRC's acceptance criteria are based on (1) draft GDC-1, insofar as it requires those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; and (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of an LOCA. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5.

Technical Evaluation

Steam Dryer

In its submittal dated July 26, 2006, TVA provided GE Report GE-NE-0000-0053-7413-R2-P, *Browns Ferry Nuclear Plant Units 1, 2, and 3 Steam Dryer Stress, Dynamic, and Fatigue Analysis for EPU Conditions*. As discussed in the GE report, the function of the steam dryer is to remove any remaining liquid in the steam exiting from the steam separators after leaving the reactor core region. The wet steam flows upward from the steam separators into an inlet plenum, horizontally through the dryer vane banks, vertically into an outlet plenum and the RPV dome, and then into the four MSLs to reach the turbine generator. The steam dryer in each unit has an active vane height of 72 in. The steam dryers are welded assemblies constructed of Type 304 stainless steel. The weld heat affected zone material may be sensitized during the fabrication process such that the steam dryers are susceptible to ISGCC.

The GE Report indicates that the Unit 1 steam dryer is a passive, nonsafety related component that was included in Class I seismic analyses. The steam dryer assembly is classified as an "internal structure" per ASME Code, Section III, Subsection NG. The steam dryer is only analyzed for load conditions for which loss of structural integrity of the steam dryer could interfere with the required performance of safety class equipment due to generation of loose parts that might prevent closure of the MSIVs or affect the core support structure integrity (shroud, top guide, core support, and shroud support).

In its submittal dated September 22, 2006, TVA states that, after receiving the power uprate license amendment, Unit 1 will proceed to operation at 105-percent OLTP. At that power level, TVA will collect plant data and perform an analysis of the steam dryer. TVA indicates that the power ascension will be controlled under the Extended Power Uprate Startup Test Program. TVA will submit the results of the steam dryer analysis to the NRC with a request for approval of operation of Unit 1 up to 120-percent OLTP.

Units 2 and 3 were uprated to 105-percent OLTP in 1998. GE Report GE-NE-0000-0053-7413-R2-P summarizes steam dryer operating experience at BFN. In the past, all three units experienced drain channel cracking with the welds subsequently reinforced to reduce the stress at those locations. Unit 3 also experienced tie bar cracking such that the tie bars were replaced with a modified design. At Unit 1, the drain channel welds will be reinforced and the modified tie bar design will be installed prior to restart.

Confidence in the capability of the steam dryer in Unit 1 to maintain its structural integrity at 105-percent OLTP conditions is based on the similarity of the three units, and the successful operating experience of Units 2 and 3 at 105-percent OLTP conditions since 1998. In its submittal dated July 26, 2006, TVA described the similarity of the BFN units and the small differences between their MS systems in response to an NRC staff request for additional information (RAI). The similarity of the three units allows the 6-year operation of Units 2 and 3 at 105-percent OLTP without significant adverse flow effects on their steam dryers to be considered in evaluating the 105-percent OLTP request for Unit 1. Further, in its submittal dated September 22, 2006, TVA indicated that the Unit 1 steam flow of 129 ft per second (fps) at current licensed conditions will increase to 132 fps at 105-percent OLTP conditions. Also, in the September 22 submittal, TVA reported that its analyses predict that resonance in the MS system from the MSRVs will not occur until the steam flow approaches the steam velocity at EPU conditions (153 fps).

In its submittal dated July 26, 2006, TVA described the Power Ascension Procedure in response to an NRC request for information. Among the plant parameters to be monitored during power ascension, TVA stated that it would obtain measurements of MSL dynamic pressure fluctuations hourly and at least once every 2.5 percent OLTP power step to evaluate the pressure loading on the Unit 1 steam dryer. TVA will also determine moisture carryover every 24 hours to provide confirmation of the continued structural integrity of the steam dryer. TVA has established performance criteria and required actions based on moisture carryover and MSL pressure spectra data during power ascension.

Conclusion

Based on the above discussion, the NRC staff has reasonable assurance regarding the structural integrity of the Unit 1 steam dryer during plant operation at 105-percent OLTP.

Steam, Feedwater, and Condensate Systems and Components

Section 3.4.3, Piping Flow Induced Vibration, of the PUSAR, stated that the increased flow associated with the proposed power uprate will result in higher vibration levels in some plant systems and their components. To address this power uprate condition, TVA stated that vibration data will be collected and evaluated for high-energy piping systems during initial power uprate operation. Section 10.4.3, Main Steam Line, Feedwater and Reactor Recirculation Piping Flow Induced Vibration Testing, of the PUSAR discussed the plans for vibration monitoring during initial plant operation at power uprate conditions. The licensee stated that vibration data will be evaluated using acceptance criteria in accordance with the ASME operation and maintenance guideline for piping steady-state vibration monitoring and evaluation.

In a letter dated July 26, 2006, TVA described the Power Ascension Procedure that will include evaluation of the performance of the steam, FW, and condensate systems and their components during power ascension at Unit 1. For example, the parameters to be monitored include moisture carryover, reactor power and rod pattern adjustments, core flow, core inlet sub-cooling, reactor water level, individual MSL flows and MSL flow element pressure data, total FW flow, and CRD flow. TVA also stated that it would obtain measurements of MSL dynamic pressure fluctuations. In evaluating plant data, TVA will monitor (1) moisture carryover every 24 hours; (2) MSL pressure hourly and at least once every 2.5-percent OLTP power step; and (3) MSL acceleration at least once every 2.5-percent power step above OLTP and within one hour after achieving every 2.5-percent power step above OLTP. TVA will perform inspections and walkdowns of the steam, FW, and condensate systems to evaluate equipment performance, and to identify the presence of abnormal vibration effects and abnormal noises or signs of deteriorating material condition. TVA has established performance criteria and required actions based on moisture carryover and MSL pressure spectra data.

Section 3.5.3 of the PUSAR indicated that the safety-related thermowells and sample probes in the piping for the MS, FW, and RCSs have been evaluated and determined to be adequate for the increased flows. In the July 26, 2006, letter, the licensee described its evaluation and technical basis for this determination. In particular, the licensee generated a finite element model of the thermowell or sample probe to calculate the component natural frequency and mode shapes. The licensee then checked the vortex shedding frequency against the component natural frequency, and determined whether the vortex-shedding frequency locks-in with the natural frequency. Using the finite element model, the licensee determined that the resulting stress in the thermowells and sample probes was less than the fatigue allowable of 13,600 psi.

Based on its review, the NRC staff considers that TVA has established acceptable plans to monitor the performance of the steam, FW, and condensate systems, and their components. Therefore, the NRC staff does not have a safety concern regarding the licensee's planned evaluation of the performance of the MS, FW, and condensate systems, and their components, during operation of Unit 1 up to and including 105-percent OLTP.

Conclusion

The NRC staff has reviewed the licensee's consideration of potential adverse flow effects on the MS, FW, and condensate systems and their components (including the steam dryer) for operation of Unit 1 at 105-percent power uprate conditions. The NRC staff concludes that the licensee has demonstrated that the MS, FW, and condensate systems and their components (including the steam dryer) at Unit 1 will continue to meet the provisions of draft GDC-1, 2, 40, and 42 following implementation of the proposed 105-percent power uprate, subject to the license condition discussed in Section 2.12. Therefore, the NRC staff finds TVA's evaluation of potential adverse flow effects to be acceptable with respect to operation at 105-percent power uprate conditions.

2.2.7 Environmental Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

Environmental qualification (EQ) of mechanical and electrical equipment involves demonstrating that the equipment is capable of performing their safety functions under significant environmental stresses which could result from design DBAs. The NRC staff's review focused on the effects of the proposed power uprate on the environmental conditions that the mechanical and electrical equipment will be exposed to during normal operation, AOOs, and accidents. The NRC staff's review was conducted to ensure that the equipment will continue to be capable of performing their safety functions following implementation of the proposed power uprate. The NRC's acceptance criteria for EQ of mechanical equipment are based on the relevant requirements set forth in 10 CFR Part 50. Specific review criteria are contained in SRP Section 3.11.

Technical Evaluation

In the June 28, 2004, request, TVA addressed the EQ of mechanical equipment at Unit 1 in Section 10.3 of the PUSAR. TVA indicated that the changes to normal and post-accident ambient conditions for mechanical equipment are addressed the same as for electrical equipment. TVA evaluated mechanical equipment with nonmetallic components that could be potentially affected by power uprate conditions and determined that the functional capability of the nonmetallic components in mechanical equipment inside or outside containment was not adversely impacted. With respect to design qualification of mechanical components, TVA stated that the process fluid operating conditions for equipment in some systems would be affected by uprated operation due to slightly increased temperatures, pressure, or flow. TVA determined that the effects of these increased loads on the EQ of mechanical equipment were not significant.

Appendices A and B of 10 CFR Part 50, provide general requirements related to EQ of mechanical equipment. In particular, components must be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs. Measures must be established for the selection and review of the suitability of materials, parts, and equipment that are essential to safety-related functions. Design control measures must be established for verifying the adequacy of design. Equipment qualification records must be maintained and include the results of tests and materials analyses.

For the EQ of mechanical equipment, the NRC staff focused its review on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). Mechanical equipment experiences the same environmental conditions as those defined in 10 CFR 50.49 for electrical equipment.

In Section 2.3.1, the NRC staff describes its evaluation of the capability of electrical equipment to continue to perform their safety functions under power uprate conditions. In that section, the NRC staff found that the licensee had adequately addressed the effects of the proposed power uprate on the EQ of electrical equipment. The NRC staff finds that the conditions used by the licensee in reviewing the EQ of electrical equipment are sufficient for the EQ of mechanical equipment in support of the proposed 105-percent power uprate.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed power uprate on the EQ of mechanical equipment at Unit 1. The NRC staff concludes that the licensee has adequately addressed the EQ of mechanical equipment for the proposed 105-percent power uprate. Therefore, the NRC staff finds the EQ of mechanical equipment to be acceptable with respect to the proposed 105-percent power uprate.

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

Regulatory Evaluation

The EQ of electrical equipment involves demonstrating that the equipment is capable of performing its safety function under significant environmental stresses which could result from DBAs. The NRC staff's review focused on the effects of the proposed power uprate on the environmental conditions that the electrical equipment will be exposed to during normal operation, AOOs, and accidents. The NRC staff's review was conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed power uprate. The NRC's acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49, which sets forth requirements for the qualification of electrical equipment important to safety that is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

Technical Evaluation

For power uprate, the licensee has evaluated the Equipment Qualification Data Packages 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, and (4) humidity. That effort established environmental profiles at uprated conditions, and either validated the existing qualification of Unit 1 equipment and instrumentation, or identified the need for replacement. Any required replacements are being performed as part of ongoing Unit 1 restart activities.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed power uprate on the EQ of electrical equipment and concludes that the licensee has adequately addressed the effects of the proposed power uprate on the environmental conditions for and the qualification of electrical equipment. The NRC staff further concludes that the electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the EQ of electrical equipment.

2.3.2 Offsite Power System

Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the offsite power system, and the stability studies for the electrical transmission grid. The NRC staff focused its review on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power (LOOP) to the plant following implementation of the proposed uprate. The NRC's acceptance criteria for offsite power systems are based on GDC-17. Specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to SRP Section 8.2, and Branch Technical Positions (BTPs) PSB-1 and ICSB-11.

Technical Evaluation

Grid Stability

For power uprate operation, the grid adequacy and stability study, provided in Enclosure 11 of the PUSAR, credits a capability of +360/-150 mega volt amps reactive for Unit 1 as the basis for analyzing the adequacy of the BFN to grid interface. TVA is an integrated utility where the transmission system is owned and operated by its owner-operated grid operations known as the Power Systems Operations (PSO) group. PSO manages planned transmission line outage schedules and establishes minimum voltage levels based on system loading and configuration. The licensee states that it has coordinated with the PSO transmission planning organization with respect to the proposed power level, post-trip plant loading data, generator capabilities, and minimum switchyard voltage acceptance criteria. PSO evaluated a range of grid conditions and identified bounding parameters (e.g., lines and transformers in service, system loading, and voltage levels) that should ensure the ability of the grid to meet the minimum switchyard voltage requirements during a unit trip with a postulated design basis event. PSO also investigated the need for system enhancements, including additional static and dynamic reactive sources in the region around the BFN plant, resulting from the restart of Unit 1 at uprated conditions. PSO concluded that at the post-uprate generator reactive output levels, no additional reactive sources were required to meet minimum post-trip voltage requirements, and that the post-uprate reactive contribution to the grid from the generators was sufficient.

Additionally, studies were performed to determine the effects of restart of Unit 1 on the adequacy of the TVA transmission system to provide GDC-17 required offsite power. Analysis of the studies determined that operation at the 105-percent electrical output will not have an adverse effect on the reliability of the offsite electrical system or on the stability of the units. The offsite power system will continue to meet GDC-17 requirements. The NRC staff has reviewed the above grid adequacy and stability studies and concludes that the proposed uprate will have no adverse impact on grid reliability.

The NRC staff has reviewed the licensee's assessment of the effects of the proposed uprate on the offsite power system and concludes that the offsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed uprate. The offsite power system has the capacity and capability to supply power to all safety loads and other required equipment. The NRC staff also concludes that the grid stability studies have demonstrated that for at 105-percent uprate conditions, the grid remains stable. Therefore, the NRC staff finds the offsite power system acceptable for operation at uprated conditions.

Main Generator

The licensee states that the Unit 1 generator has been rewound and uprated to 1330 mega volt amps (MVA). For uprate conditions, the generator hydrogen operating pressure for Unit 1 will be increased to a design pressure rating of 75 psig. The hydrogen pressure regulators and associated setpoints will be adjusted for 75 psig operation.

The NRC staff concludes that by adjusting the generator hydrogen operating pressure from 65 psig to 75 psig for extra cooling, the main generator operation will be acceptable at 1330 MVA after hydrogen pressure regulators and associated setpoints are adjusted for 75 psig operation for uprated conditions.

Iso-Phase Bus Duct

The iso-phase bus operates at 22 kV. The bus is divided into several sections with rating appropriate for each section. The main phase bus duct for all units is to be modified to have a continuous rating of 36,740 amperes from the present rating of 35,270 amperes. The generator bus is to be modified to have a continuous rating of 18,370 amperes from the current 17,635 amperes. The delta bus will be modified to have a continuous rating of 21,212 amperes from the current 20,365 amperes. The modification will include replacement of the cooling coil with a higher capacity coil, replacement of the single cooling fan with dual cooling fans, duct work modifications, damper replacements and changes to instrumentation and controls. These modifications for iso-phase bus will be in place for 105-percent power uprate operation for Unit 1.

The NRC staff has reviewed the modifications to the iso-phase bus and concludes that the iso-phase bus will be acceptable after modifications to accommodate the extra heat loads at 105-percent power uprate operation.

Main Bank Transformer

The licensee states that the Unit 1 main bank transformer (MBT) has been replaced as a material improvement due to aging and reliability concerns. MBT cooling equipment and MBT high and low voltage winding connection hardware have also been replaced. The new transformer is rated at 500 MVA at 65 degrees Celsius per phase and is adequate to support Unit 1 operation at uprated conditions.

The NRC staff concludes that by increasing the MBT rating from the current 448 MVA to 500 MVA, Unit 1 MBTs will be able to carry the main generator loading under uprated conditions.

Unit Auxiliary/Startup Transformers

The licensee's evaluation confirms that the current ratings of unit auxiliary/startup transformers are adequate to support Unit 1 operation at uprated conditions. The NRC staff reviewed the proposed ratings for the unit auxiliary/startup transformers contained in Table EEIB-B.4-4 of the licensee's December 19, 2005, submittal. As the current licensed thermal power (CLTP) transformer ratings remain adequate for uprated conditions and the total calculated loading on the unit auxiliary/startup transformers are acceptable under uprated conditions, the NRC staff finds the unit auxiliary/startup transformers are acceptable.

Non-Class 1E Loads

The licensee states that no modification to the reactor recirculation pumps, condensate pumps, and condensate booster pumps will be required for the 105-percent operating conditions. The NRC staff concludes that since the extra heat load at 105 percent is within the ratings of the reactor recirculation pumps, condensate pumps, and condensate booster pumps, they are acceptable for the proposed 105-percent uprate.

Conclusion

Based on its review, the NRC staff concludes that the offsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed uprate. Also, the impact of the proposed uprate does not degrade grid stability. Grid stability studies have demonstrated that for power uprate operation the transmission grid remains stable. Therefore, the offsite power system is acceptable for operation at power uprate conditions.

2.3.3 Emergency Diesel Generators

Regulatory Evaluation

The ac onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to the safety-related equipment. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the ac onsite power system. Acceptance criteria are based on GDC-17 as it relates to the capability of the ac onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Technical Evaluation

The NRC staff reviewed the licensee's submittal to determine whether the EDGs would remain capable of performing their intended design function at uprated conditions. The licensee stated that its review of the loads for operation at uprated conditions indicates that there is no increase in flow or pressure is required of any ECCS equipment for power uprate operation. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) is not increased with EPU, and the current EDGs loading analysis remain acceptable for power uprate operation. The EDGs have sufficient capacity to supply all required loads to achieve and maintain safe shutdown conditions to operate the ECCS equipment following postulated accidents and transients. As such, no EDG modifications are required to support Power uprate operation.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed uprate on the ac onsite power system and concludes that since there are no changes to the safety-related loads, the capacity of each EDG is adequate to support the operation under uprated conditions and no EDG modifications are required to support power uprate operation. The NRC staff further concludes that the ac onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed uprate. Therefore, the NRC staff finds the onsite ac power system acceptable for operation at uprated conditions.

2.3.4 Direct Current Power System

Regulatory Evaluation

The direct current (dc) power system includes those dc power sources and their distribution systems and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment. The NRC staff's review covers the information, analyses, and referenced documents for the dc onsite power system. Acceptance criteria are based on GDC-17 as it relates to the capability of the dc onsite electrical power to facilitate the functioning of SSCs important to safety. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2.

Technical Evaluation

The NRC staff reviewed the licensees' submittal to determine whether the dc system and its components would remain capable of performing their intended design function at uprated conditions. The licensee states that there is no impact from power uprate on the safety-related batteries; therefore, an evaluation of the dc power system is not required.

2.3.5 Station Blackout

Regulatory Evaluation

Station blackout (SBO) refers to a complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant, and involves a LOOP concurrent with turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources." The NRC staff focused its review on the impact of the proposed uprate on the plant's ability to cope with and recovery from an SBO for the period of time established in the plant's licensing basis. The NRC's acceptance criteria for SBO are based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and Appendix B to SRP 8.2.

Technical Evaluation

For power uprate, the licensee reevaluated SBO using the guidelines of NUMARC 87-00. The plant response to and coping capabilities for an SBO event are affected slightly by operation at uprated conditions, due to the increase in the initial power level and decay heat. The licensee states that there are no changes to the systems and equipment used to respond to an SBO and that the Unit 1 SBO coping duration of 4 hours has not changed under uprated conditions.

Areas containing equipment necessary to cope with an SBO event were evaluated for the effect of loss-of-ventilation due to an SBO. The evaluation shows that equipment operability is bounded due to conservatism in the existing design and qualification bases. The battery capacity remains adequate to support HPCI/RCIC pump operation at uprated conditions. Also, adequate compressed gas capacity exists to support the MSRV actuations.

The current condensate storage tank (CST) inventory reserve (135,000 gallons), for HPCI/RCIC use, ensures that adequate water is available to remove heat, depressurize the reactor, and maintain RV above top of active fuel (approximately 122,000 gallons required).

Based on the above, the NRC staff concludes that since there are no changes to the systems and equipment used to respond to an SBO and the equipment operability is bounded due to conservatism in the existing design, the SBO will be unaffected by uprate to 105 percent.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed uprate on plant's ability to cope with recovering from an SBO for the period of time established in the plant's licensing basis. The NRC staff concludes that the licensee has adequately evaluated the effects of the proposed uprate on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following the implementation of the proposed uprate.

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

Instrumentation and control systems are provided (1) to control plant processes having a significant impact on plant safety; (2) to initiate the reactivity control system (including control rods); (3) to initiate the ESF systems and essential auxiliary supporting systems; and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems. The NRC staff conducted a review of the reactor trip system, ESF actuation system (ESFAS), safe shutdown systems, control systems, and diverse instrumentation and control systems for the proposed power uprate to ensure that they are adequately designed to meet their safety functions. The NRC staff's review was also conducted to ensure that failures of the systems do not affect safety functions. The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and draft GDC-1, 11, 12, 14, 15, 19, 20, 22, 23, 25, 26, 40, and 42. Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

Technical Evaluation

For the proposed power uprate, the licensee evaluated each existing instrument of the affected nuclear steam supply systems (NSSS) and BOP systems to determine their suitability for the revised operating range of the affected process parameters. Where operation at the power uprate condition impacted safety analysis limits, the licensee verified that the acceptable safety margin continued to exist under all conditions of the power uprate. Where necessary, the licensee revised the setpoint and uncertainty calculations for the affected instruments. Since Unit 1 is restarting after a long shutdown, the licensee has modified many systems and components to meet the NRC requirements or because of obsolescence. In its letter of August 15, 2005, the licensee identified these modifications and identified similar modifications were performed on Units 2 and 3. In its letter, the licensee has also identified the restart test program and power ascension test program which will verify the proper operability of these systems and components. The adequacy of the restart test program and power ascension test program is reviewed by the NRC staff and discussed in Section 2.12 of this SE.

In addition, the licensee is planning changes to accommodate the revised process parameters. These changes are based on system analyses reviewed by the NRC staff. The licensee will confirm the acceptability of these changes during power ascension testing. The NRC staff finds that upon completion of the modifications, the Unit 1 instrumentation and control system should accommodate the proposed power uprate without compromising safety. The above changes do not effect the licensee's compliance with the existing plant licensing bases.

In a separate submittal dated January 10, 2006, as supplemented by letters dated April 14, August 1, September 5 and 14, 2006, the licensee identified that instrument setpoints in the TS are established using the setpoint methodology discussed in these submittals. This setpoint methodology was reviewed by the NRC staff and found acceptable in a letter dated September 14, 2006. The NRC staff therefore, finds this setpoint methodology acceptable in determining new setpoints proposed by the licensee for the power uprate application.

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

Conclusion

The NRC staff has reviewed the licensee's application related to the effects of the proposed power uprate on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. The NRC staff concludes that the licensee has adequately addressed the effects of the proposed power uprate on these systems and that the changes that are necessary to achieve the proposed power uprate are consistent with the plant's design basis. The NRC staff further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and draft GDC-1, 11, 12, 14, 15, 19, 20, 22, 23, 25, 26, 40, and 42. Therefore, the NRC staff finds the instrumentation and controls acceptable for operation at uprated conditions.

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

For proposed power uprates, the NRC staff reviews flood protection measures to ensure that SSCs important to safety are adequately protected from the consequences of internal flooding that result from postulated failures of tanks and vessels. Because the NRC staff's review focuses on increases of fluid volumes in tanks and vessels that will occur as a result of a power uprate and the licensee has indicated that the proposed power uprate does not result in an increase in such fluid volumes at Unit 1, an evaluation of this particular area by the NRC staff is not required.

2.5.1.1.2 Equipment and Floor Drains

The function of the equipment and floor drainage system (EFDS) is to assure that waste liquids, valve and pump leak-offs, and tank drains are directed to the proper area for processing or disposal while preventing a backflow of water that might result from maximum flood levels to areas of the plant containing equipment that is important to safety. The EFDS also protects against the potential for inadvertent transfer of contaminated fluids to an uncontaminated drainage system. The licensee indicated in the PUSAR that the EFDS for Unit 1 is not impacted by the proposed power uprate. Therefore, an evaluation of the EFDS is not required.

2.5.1.1.3 Circulating Water System

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove excess heat from the turbine cycle and auxiliary systems. For proposed power uprates, the NRC staff's review of the CWS focuses on the impact that the proposed uprate will have on existing flooding analyses due to any increases that may be necessary in fluid volumes and installation of larger capacity CWS pumps or piping. Because the impact of the proposed power uprate on the licensee's flooding analysis is considered in Sections 2.5.1.1.1 and 2.5.1.3 of this evaluation, a separate evaluation for the CWS in this section is not required.

2.5.1.2 Missile Protection

2.5.1.2.1 Internally Generated Missiles

Regulatory Evaluation

The NRC staff's review concerns missiles that could result from in-plant component overspeed failures and high-pressure system ruptures. The NRC staff's review of potential missile sources covered pressurized components and systems, and high-speed rotating machinery. The NRC staff's review was conducted to ensure that safety-related SSCs are adequately protected from internally generated missiles. In addition, for cases where safety-related SSCs are located in areas containing non-safety-related SSCs, the NRC staff reviewed the non-safety-related SSCs to ensure that their failure will not preclude the intended safety function of the safety-related SSCs. The NRC staff's review focused on any increases in system pressures or component overspeed conditions that could result during plant operation, AOOs, or changes in existing system configurations such that missile barrier considerations could be affected. The NRC's acceptance criteria for the protection of SSCs important to safety against the effects of internally generated missiles that may result from equipment failures, are based on draft GDC-40. Specific review criteria are contained in SRP Sections 3.5.1.1 and 3.5.1.2.

Technical Evaluation

The licensee evaluated the impact of the proposed uprate on SSCs important to safety due to internally generated missiles that may result from failures in high energy systems and overspeed of rotating equipment in a letter dated February 23, 2005. In response to questions, the licensee provided additional information supporting its conclusion that the consequences of internally generated missiles will not be affected by the proposed uprate. Specifically, the impact associated with replacement of the FW pumps, and the increased FW flow due to uprate operation was addressed. Since the new FW pumps will be oriented in the same direction as the current FW pumps and will continue to use the existing suction and discharge connections, the existing missile evaluation will remain valid. Furthermore, the pumps are located in the turbine building where no safety related SSCs are located. However, FW piping going from the turbine building to the reactor building pass in close proximity to safety-related SSCs. Missile barriers and pipe whip restraints on high energy lines currently provide protection for the affected SSCs. Since the designs of the restraints and missile shields are based on FW system design pressure (as opposed to system operating pressure), and the design pressure is not changed for uprate, the licensee concluded that SSCs important to safety will continue to be protected from plant internally generated missiles.

The licensee also provided additional information to address the impact of uprate on the consequences of main turbine missiles that could be generated. TVA indicated that the three turbines are separately housed in an adjacent turbine building and all three turbines are laid out in parallel and rotate on an axis that is perpendicular to the reactor building. The orientation results in the main turbine being categorized as a “favorable” orientation with regard to missile failure probability analyses, and the licensee confirmed that the turbine missile criteria for Unit 1 will continue to be satisfied following uprate implementation.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the potential impact of the proposed power uprate on existing considerations and features that are credited for protecting equipment important to safety from the effects of internally generated missiles. The licensee has determined that the uprate will not cause the effects of internally generated missiles (outside containment) on SSCs important to safety to be more severe than previously assumed and therefore, the NRC staff agrees that SSCs important to safety will continue to be adequately protected from internally generated missiles following uprate implementation.

Conclusion

The NRC staff has reviewed the changes in system pressures and configurations that are required for the proposed uprate and concludes that SSCs important to safety will continue to be protected from internally generated missiles and will continue to meet the provisions of draft GDC-40 following implementation of the proposed uprate. Therefore, the NRC staff finds the evaluation of internally generated missiles acceptable for power uprate conditions.

2.5.1.2.2 Turbine Generator

Regulatory Evaluation

The turbine control system, steam inlet stop and control valves, low pressure turbine steam intercept and inlet control valves, and extraction steam control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant. The NRC staff's review of the turbine generator focused on the effects of the proposed power uprate on the turbine overspeed protection features to ensure that a turbine overspeed condition above the design overspeed is very unlikely. The NRC's acceptance criteria for the turbine generator are based on draft GDC-40, and relates to protection of ESFs from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles. Specific review criteria are contained in SRP Section 10.2.

Technical Evaluation

The Unit 1 main turbine has been modified to accommodate the increased steam flow for EPU operation. The modifications include use of monoblock rotors, new (heavier) buckets and new diaphragms. The monoblock rotors and new buckets increase the rotor inertia which tends to slow the acceleration rate of the turbine upon a loss of load event. However, the entrapped steam energy contained within the turbine and associated piping tends to increase the acceleration rate of the turbine upon a loss-of-load event. Emergency overspeed (EOS) protection for the main turbine is an independent two-out-of-three logic electronic trip device with independent speed sensors that detect turbine speed and send a trip signal to the master trip solenoid valve (MTSV) when the turbine speed exceeds the EOS setpoint. Electronic turbine overspeed protection is also provided by the digital electro-hydraulic control (EHC) system, which receives turbine speed input from its own independent speed sensors and sends a separate trip signal to the MTSV upon detecting a turbine overspeed condition. The EOS turbine trip setpoint is established based on a GE criterion that limits the turbine speed to 120-percent of the turbine rated speed; and the digital EHC turbine trip setpoint is maintained less than or equal to the EOS setpoint.

The impact that EPU will have on TG overspeed protection is discussed in Section 7.1 of the Unit 1 PUSAR. Also, in a letter from TVA dated June 15, 2006, the licensee provided additional information related to turbine overspeed protection. The licensee evaluated the impact of EPU on the emergency (most limiting) turbine overspeed scenario, where it is assumed that the turbine EHC system and the turbine control and intercept valves fail to respond to the initial turbine speed increase following the most limiting load rejection event. For this most limiting scenario, the turbine rapidly accelerates to the EOS trip setpoint, where the turbine EOS electronic overspeed trip device causes the main and intermediate stop valves to trip. Based on its evaluation, the licensee determined that an EOS turbine trip setpoint of 109.5-percent of rated speed will limit turbine speed overshoot to 119-percent of rated speed which satisfies the GE criterion. The licensee indicated that the EOS setpoint for the Unit 1 turbine would be set at 109-percent of turbine rated speed (same as the setpoint for the Units 2 and 3 turbines), which provides additional margin. The licensee also indicated that, a) turbine overspeed functional testing will be conducted at no load conditions as part of the startup test program; b) an EOS trip logic functional test will be conducted to validate (through the insertion of simulated speed signals) that the overspeed trip signal to the MTSV is received at the specified EOS trip setpoint; c) the turbine trip output from the EHC digital control system will also be tested for proper trip signal initiation to the MTSV; and d) the MTSV is functionally tested on a weekly basis.

Based on its review, the NRC staff finds that the licensee has adequately evaluated and addressed the potential impact of the proposed power uprate on the capability to prevent turbine overspeed. The licensee has established the EOS trip setpoint consistent with the criterion that is specified by GE, the digital EHC turbine trip setpoint will be maintained less than or equal to the specified EOS turbine trip setpoint, and proposed testing is sufficient to assure proper performance. Therefore, the NRC staff agrees that the turbine overspeed protective features will continue to prevent turbine overspeed consistent with the turbine design criteria and therefore, the NRC staff finds that the power uprate will not increase the likelihood that turbine missiles will be generated due to turbine overspeed conditions.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed power uprate on the turbine generator and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on turbine overspeed. The NRC staff concludes that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the guidance of draft GDC-40 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the turbine generator.

2.5.1.3 Pipe Failures

Regulatory Evaluation

The NRC staff conducted a review of the plant design for protection from piping failures outside containment to ensure that (1) such failures would not cause the loss of needed functions of safety-related systems and (2) the plant could be safely shut down in the event of such failures. The NRC staff's review of pipe failures included high and moderate energy fluid system piping located outside of containment. The NRC staff's review focused on the effects of pipe failures on plant environmental conditions, control room habitability, and access to areas important to safe control of post-accident operations where the consequences are not bounded by previous analyses. The NRC's acceptance criteria for pipe failures are based on draft GDC-40 and 42, insofar that they require that ESFs be designed to accommodate the dynamic effects of postulated pipe ruptures, as well as the effects of an LOCA. Specific review criteria are contained in SRP Section 3.6.1.

Technical Evaluation

The uprated plant requires a small (less than 3 percent) increase in RPV dome pressure to supply sufficient steam to the main turbine for operating at EPU conditions. The slight increase in vessel pressure and temperature will result in a small increase in the mass and energy release rates following postulated pipe failures. The licensee's evaluation of the impact that EPU will have on the consequences of high and moderate energy piping failures located outside containment is discussed in Sections 10.1 and 10.2 of the PUSAR. Also, the licensee indicated in a letter dated July 26, 2006, that no new break locations in the main steam and feedwater piping are required to be postulated due to the proposed power uprate. TVA also indicated that plant walk-downs will be performed to confirm that pipe whip restraints have been installed with no significant changes in configuration from what was specified on the original drawings.

The licensee's evaluation of internal flooding due to high energy line breaks is addressed in Section 10.1.3 of the PUSAR. The licensee determined that the reactor water cleanup (RWCU) and the reactor feedwater systems are the only two high-energy systems with liquid filled lines that pose a flooding concern. The licensee indicated in a letter dated February 23, 2005, that the proposed EPU will not result in any significant changes (less than 1-inch) in internal flooding levels and that mechanical equipment will not be prevented from performing necessary safety-related functions. The licensee's conclusion is based on evaluation of both high and moderate energy pipe failures. Also, in a letter dated June 7, 2006, the NRC staff previously evaluated and accepted the licensee's moderate energy pipe failure flooding analysis, which included consideration of the proposed EPU.

Based on a review of the information that was provided, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of the proposed power uprate on the consequences of postulated high and moderate energy pipe failures, including flooding considerations. The licensee determined that the proposed power uprate will not result in any new pipe failure locations, and the consequences of postulated pipe failures will not exceed plant design limitations that were previously recognized and credited. Therefore, the staff agrees that the capability to mitigate postulated pipe failures in accordance with licensing-basis considerations will not be compromised by operating at the proposed EPU power level.

The licensee has not requested NRC review and approve of any changes to the licensing basis related to pipe failure for EPU operation; and this evaluation does not constitute NRC approval of any changes that are being made to the licensing basis in this regard.

Conclusion

The NRC staff has reviewed the changes that are necessary for the proposed power uprate and the licensee's proposed operation of the plant, and concludes that SSCs important to safety will continue to be protected from the dynamic effects of postulated piping failures in fluid systems outside containment and will continue to meet the provisions of draft GDC-40 and 42 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to protection against postulated piping failures in fluid systems outside containment.

2.5.1.4 Fire Protection

Regulatory Evaluation

The purpose of the fire protection program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that SSCs required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe shutdown following a fire. The NRC's acceptance criteria for the FPP are based on

(1) 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50, insofar as they require the development of an FPP to ensure, among other things, the capability to safely shut down the plant; (2) draft GDC-3, insofar as it requires that the reactor facility be designed (a) to minimize the probability of events, such as fire and explosions, and (b) to minimize the potential effects of such events to safety; and (3) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing. Specific review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

In Attachment 1 to Matrix 5 of RS-001, Supplemental Fire Protection Review Criteria, it is stated that:

> . . . power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire . . . . [W]here licensees rely on less than full capability systems for fire events . . ., the licensee should provide specific analyses for fire events that demonstrate that (1) fuel integrity is maintained by demonstrating that the fuel design limits are not exceeded and (2) there are no adverse consequences on the reactor pressure vessel integrity or the attached piping. Plants that rely on alternative/dedicated or backup shutdown capability for post-fire safe shutdown should analyze the impact of the power uprate on the alternative/dedicated or backup shutdown capability . . . . The licensee should identify the impact of the power uprate on the plant's post-fire safe shutdown procedures.

Section 6.7, Fire Protection, of the PUSAR addresses the Fire Protection provisions of RS-001. As Unit 1 has been shutdown and defueled since 1985, the licensee is relying on administrative controls, procedures and resources provided for Units 2 and 3. The results of the Appendix R evaluation demonstrate that fuel cladding integrity and containment integrity are maintained. During an inspection in September 2006, the NRC reviewed the licensee's fire protection procedures for Unit 1. The inspection staff found that an evaluation of the time needed to perform operator manual actions had been satisfactorily completed. However, as the safe shutdown procedures for Unit 1 are currently in progress, in that the feasibility to perform operator manual actions has not yet been completed, the NRC inspection staff will confirm sufficient time is available for the operator to perform the necessary actions.

Conclusion

The NRC staff has reviewed the licensee's fire-related safe shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions.

The NRC staff further concludes that the FPP will continue to meet the requirements of 10 CFR 50.48, Appendix R to 10 CFR Part 50, and draft GDC-3 and 4 following implementation of the proposed uprate. Therefore, the NRC staff finds the FPP acceptable to support operation at uprated conditions.

2.5.2 Fission Product Control

2.5.2.1 Fission Product Control Systems and Structures

The purpose of the NRC staff's review of fission product control systems and structures is to confirm that current analyses remain valid or have been revised, as appropriate, to properly reflect the proposed power uprate conditions. Consequently, the NRC staff's review focuses primarily on any adverse effects that the proposed power uprate might have on the assumptions that were used in analyses that were previously completed. Because the impact of EPU on plant systems and structures identified by the licensee as making up the fission product control system are addressed in Section 2.6, *Containment Review Considerations*, Section 2.7, *Habitability, Filtration, and Ventilation*, and Section 2.9, *Source Terms and Radiological Consequences*, a separate review of this area is not required.

2.5.2.2 Main Condenser Evacuation System

The main condenser evacuation system (MCES) is not impacted by the proposed power uprate because the condenser air removal requirements are not affected. The MCES is sized based upon the volume of the condenser and desired evacuation time, neither of which is impacted by the proposed uprate. Consequently, the existing capability to monitor the MCES effluent is also not affected by the proposed power uprate and therefore, NRC review of the MCES is not required.

2.5.2.3 Turbine Gland Sealing System

Regulatory Evaluation

The turbine gland sealing system (TGSS) is provided to control the release of radioactive material from steam in the turbine to the environment. The NRC staff reviewed changes to the turbine gland sealing system with respect to factors that may affect gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths). The NRC's acceptance criteria for the turbine gland sealing system are based on (1) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) draft GDC-17, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs and postulated accidents. Specific review criteria are contained in SRP Section 10.4.3.

Technical Evaluation

In a letter dated March 7, 2006, the licensee provided its evaluation of the TGSS. The TGSS prevents the leakage of steam into the turbine building and also prevents the leakage of air into the main condenser. During normal power operations, a pressure regulating valve and two sealing steam header unloader valves maintain the sealing steam header pressure at approximately 4 psig. In order to regulate the sealing steam header pressure, the unloader valves divert excess sealing steam to the main condenser. For EPU, the licensee is installing larger unloader valves to accommodate modifications that are being made to the main turbine. The larger unloader valves will provide the TGSS with additional capability to maintain the sealing steam pressure at 4 psig, thereby maintaining its capability to contain activated nitrogen and limit radiation discharge to the environment.

Based on a review of the information that was provided, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of the proposed power uprate on the capability of the TGSS to perform its functions. The licensee has determined that the installation of larger unloader valves will assure sufficient TGSS steam pressure to prevent leakage past the turbine glands consistent with existing turbine design specification. Therefore, the NRC staff finds that with the installation of larger unloader valves, the TGSS will continue to prevent leakage past the turbine glands.

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the turbine gland sealing system and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the turbine gland sealing system will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment consistent with draft GDC-17 and 70. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the turbine gland sealing system.

2.5.3 Component Cooling and Decay Heat Removal

2.5.3.1 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool (SFP) provides wet storage of spent fuel assemblies and temporary storage of new fuel assemblies. The safety function of the SFP cooling system (SFPCS) is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The NRC staff's review of the SFPCS for proposed power uprates focuses on the impact that proposed power uprates will have on the capability of the SFPCS to provide adequate cooling of the spent fuel during all operating and accident conditions.

Regulatory Evaluation

The criteria that are most applicable to the NRC staff's review of the SFPCS are based primarily on draft GDC-4, Sharing of Systems, insofar that reactor facilities should not share systems or components unless it is shown that safety is not impaired by the sharing; draft GDC-67, Fuel and Waste Storage Decay Heat, insofar that reliable decay heat removal systems should be designed to prevent damage to the fuel in storage; and other licensing-basis considerations that are applicable. The NRC staff's review of the SFPCS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for power uprate operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 10.5 of the UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The SFPCS has two pumps that circulate the fuel pool water through a heat exchanger and a filter demineralizer. The system also has a cross-connection with the RHR system which allows the RHR system to provide supplemental cooling of the SFP. Additionally, as part of the restart activities for Unit 1, the Auxiliary Decay Heat Removal System (ADHRS), which can be used to remove residual heat from the SFP and reactor cavity during outages, will be extended to Unit 1. The SFPCS, including supplemental fuel pool cooling and the ADHRS, are nonsafety systems. To ensure adequate makeup under all normal and off normal conditions, the RHR/residual heat removal service water (RHRSW) connection provides a permanently installed seismic Category 1 qualified makeup water source for the SFP. This ensures that irradiated fuel is maintained, submerged in water, and that reestablishment of normal fuel pool water level is possible under all anticipated conditions.

The current licensing basis for the fuel pool cooling system is to maintain the SFP bulk water temperature at or below 125 degrees F for a normal batch off-load (approximately 332 fuel bundles) and below 150 degrees F for abnormal (full core) offload conditions (UFSAR, Section 10.5.5). The limiting condition is a full core discharge with all remaining storage locations filled with used fuel from prior discharges. As a result of the proposed power uprate, the normal and abnormal SFP heat loads will be higher than the pre-uprate heat loads. Power uprate will result in higher decay heat in the discharged bundles to the SFP as well as an increase in the number of discharged fuel bundles at the end of each cycle.

To ensure adequate SFP cooling for power uprate conditions, the licensee performed analyses for both, batch and full core offload scenarios based on plant operation at the proposed EPU level of 3952 MWt. In Table 6-3 of Section 6.3.1 of the PUSAR, the licensee presented the results for two system configurations, demonstrating that for each configuration the existing systems have adequate SFP cooling capability for both fuel off-load cases. In Configuration 1, the licensee used one train each of the SFPCS and the ADHRS. In Configuration 2, the licensee's analysis is based on one train each of SFPCS and the RHR system. Based on the information provided in Table 6-3, the bulk pool temperature for the limiting full core off-load and normal batch offload will remain within the design capacity of the SFPCSs.

The licensee currently performs a cycle-specific analyses prior to each offload. After power uprate, to ensure adequate SFP cooling capability, the licensee will continue to perform cycle-specific calculations to ensure the fuel pool heat load does not exceed the available cooling capacity, and that the SFP temperature limits will not be exceeded after the power uprate is implemented. Additionally, the licensee has established a regulatory commitment to implement procedure changes that will (1) define and control the generation of cycle-specific fuel pool heat load calculations; and (2) control the installation of the fuel pool gates based on the calculated fuel pool heat load.

Based on the NRC staff's review, the licensee has administrative controls in place to ensure that backup cooling capability is provided for all SFP cooling scenarios, and the licensee's commitment to implement procedure changes to enhance the administrative controls that currently exist, the NRC staff finds that the licensee has adequately evaluated and addressed the potential impact of the proposed power uprate on the capability of the SFPCS to cool the spent fuel. The licensee has determined that the existing design capacity of the SFPCS will continue to exceed the SFP heat load that results from power uprate operation and the 4-hour time to boil following a loss of SFP cooling for the full core offload case will continue to afford plant operators sufficient time to take corrective actions. The licensee's analysis was performed at the EPU power level and bounds interim plant operation at the proposed 105-percent power level. Therefore, the NRC staff finds that the design-basis capability of the SFPCS will be maintained following the implementation of the proposed interim power uprate.

Conclusion

The NRC staff has reviewed the licensee's assessment related to the SFP cooling and cleanup system and concludes that the licensee has adequately accounted for the effects of the proposed uprate on the SFP cooling function of the system. Based on this review, the NRC staff concludes that the SFP cooling and cleanup system will continue to provide sufficient cooling capability to cool the SFP following implementation of the proposed power uprate and will continue to meet the provisions of draft GDC-4 and 67. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the SFP cooling and cleanup system.

2.5.3.2 Station Service Water System

Regulatory Evaluation

The station service water system (SWS) provides essential cooling for safety-related equipment and may also provide cooling for nonsafety-related auxiliary components that are used for normal plant operation. The safety objective of the SWS is to provide cooling water to systems and components that are credited for accident mitigation. The NRC staff's review of the proposed power uprates focuses on the impact that the proposed power uprate will have on the capability of the SWS to perform its safety functions. The criteria most applicable to the NRC staff's review are based primarily on draft GDC-4, Sharing of System, insofar that reactor facilities should not share systems or components unless it is shown that safety is not impaired by the sharing; draft GDC-41, insofar that the SWS is relied upon by ESFs for performing their safety functions; draft GDC-44, insofar that the SWS is relied upon by ECCSs for performing their safety functions; draft GDC-52, insofar that the SWS is relied upon by containment heat removal systems for performing their safety functions; draft GDC-67, insofar that the SWS is relied upon by fuel and waste storage decay heat removal systems for performing their functions; and other licensing-basis considerations that are applicable. The NRC staff's review of the SWS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for power uprate operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Sections 10.9 and 10.10 of the UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The safety-related service water systems at Browns Ferry consist of Emergency Equipment Cooling Water (EECW) and RHRSW systems. The EECW system removes heat from the 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. With the exception of the heat loads associated with the RHR and CS room coolers, the heat loads due to operation of the essential components that are serviced by the EECW system were found to remain unchanged for post-LOCA conditions while operating at 3952 MWt.

Because the proposed uprate results in a slight increase in room temperature (less than 2 degrees F for RHR and less than 3 degrees F for CS), heat loads for the RHR and CS room coolers will increase slightly. However, the impact on the overall heat load is not significant and the cooling capacity of the EECW system will remain adequate for plant operation during the proposed power uprate conditions.

For the RHRSW system, the licensee's containment cooling analysis indicated that the post-LOCA RHR heat load increases due to an increase in the maximum suppression pool temperature that occurs following an LOCA. The licensee evaluated the existing suppression pool structure and associated equipment based on the increased post-LOCA suppression pool temperature and based on its evaluation, the licensee concluded that the current RHRSW system has sufficient capability to maintain the suppression pool temperature within acceptable limits following an LOCA. The licensee also determined that the RHRSW system is capable of providing adequate cooling and makeup water to the SFP heat exchangers and the SFP, respectively; and that the RHRSW system has sufficient capacity to serve as a standby coolant supply for long term core and containment cooling at the proposed power uprate conditions.

The safety-related service water systems (EECW and RHRSW) are the only systems that transfer heat from safety-related SSCs to the ultimate heat sink and are within the scope of GL 89-13. The licensee has determined that no changes to the flow rates of these systems are required to support the proposed power uprate and consequently, the key heat exchanger parameters (such as fouling factors, effectiveness and tube plugging analysis) that are used in the power uprate analysis remain consistent with the existing GL 89-13 program for Unit 1 and that current evaluations, testing, and monitoring performed by the TVA Heat Exchanger Program to meet the commitments related to GL 89-13 will continue to support operation at the proposed power uprate conditions. Also, because licensing-basis considerations will continue to be satisfied, the licensee concluded that the current SWS performance capability and flow balance are sufficient for the proposed power uprate. For example: no SWS flow rate changes or supply temperatures are required to support normal, shutdown, or accident conditions; and the SWS pump required NPSH and available NPSH are not affected and all heat exchangers will continue to operate within design limitations following the proposed power uprate.

The response to GL 96-06, *Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions*, was accomplished using a peak drywell temperature of 336 degrees F, which bounds the peak drywell temperature for the interim power uprate. Based on its evaluations, the licensee found that the reactor building closed cooling water was the system most susceptible to water hammer and two-phase flow. Because the conditions that were assumed in the existing GL 96-06 evaluation bounds the proposed power uprate conditions, the licensee's GL 96-06 evaluation is not affected by the proposed power uprate.

Based on a review of the information that was submitted, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed power uprate on the capability of the SWS (including the EECW and RHRSWS) to perform its safety functions. Because design limitations of SSCs will not be exceeded and licensing-basis considerations will continue to be satisfied, the NRC staff finds that the capabilities of the SWS will not be impacted by the proposed power uprate. Additionally, existing GL 89-13 programmatic controls will continue to assure that heat exchanger performance is maintained consistent with licensing-basis considerations following implementation of the proposed power uprate, and the GL 96-06 evaluation is not affected by the proposed uprate.

Conclusion

The NRC staff has reviewed the licensee's assessment of the impact that the proposed power uprate will have on the SWS (including the EECW and the RHRSW) and finds that the SWS will continue to be capable of performing its equipment cooling and decay heat removal functions in accordance with licensing-basis considerations. Therefore, the proposed power uprate is considered to be acceptable with respect to the SWS.

2.5.3.3 Ultimate Heat Sink (UHS)

The UHS provides the cooling medium for dissipating the heat removed from the reactor and its auxiliaries during normal operation, refueling, transient, and accident conditions. The Wheeler Reservoir along with the Tennessee River serve as the UHS for the units, and because its cooling capacity far exceeds the shutdown cooling and accident heat loads for the BFN units, the UHS is unaffected by the proposed power uprate. Therefore, a detailed evaluation of the UHS is not required.

2.5.4 Balance-of-Plant Systems

2.5.4.1 Main Steam

The MS supply system (MSSS) transports steam from the reactor to the power conversion system and to various auxiliary steam loads. The NRC staff review of the MSSS for proposed power uprates focuses primarily on system modifications that are being made that could impact the performance of safety related turbine-driven pumps. Because the proposed power uprate will not result in any modifications of this nature, this area of review is not affected by the proposed power uprate. Therefore, an evaluation of the MSSS is not required.

2.5.4.2 Main Condenser

The main condenser system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine steam bypass system, and is typically credited for providing sufficient condensate retention time to allow short-lived radioactive isotopes to decay. Unit 1 does not have an MSIV leakage control system, however, the main condenser system is credited for providing holdup and plate-out of radioactive iodine through the MSIV bypass leakage pathway following core damage. The NRC staff review for proposed power uprates focused primarily on any changes that made to the MSIV bypass leakage pathway to confirm that the isolation boundary was properly established. Because the proposed power uprate will not result in any changes to the MSIV bypass leakage pathway boundaries, this area of review is not affected by the proposed power uprate. Therefore, an evaluation of the main condenser system is not required.

2.5.4.3 Turbine Steam Bypass System

The turbine steam bypass system (TSBS) is a nonsafety-related system designed to discharge a stated percentage of rated MS flow directly to the main condenser, bypassing the turbine and enabling the plant to take step-load reductions up to the capacity of the TSBS without causing the reactor or turbine to trip. The NRC staffs review for proposed power uprate focuses primarily on any modifications that are being made to the TSBS that may warrant the performance of confirmatory testing. Because changes are not being made in the design and operation of the TSBS for power uprate operation, an evaluation of the TSBS is not required.

2.5.4.4 Condensate and Feedwater

Regulatory Evaluation

The condensate and FW system (CFS) provides FW at a particular temperature, pressure, and flow rate to the reactor. While the CFS does not perform a safety function, marginal system design and operational capability could result in loss of FW transients and increased challenges to safety systems. The NRC staff's review of the CFS for proposed power uprates focuses primarily on system design limitations and reductions in operational flexibility that will result due to power uprate operation. The acceptance criteria that are most applicable to the NRC staff's review of the CFS for proposed power uprates are based on existing plant licensing-basis considerations, especially with respect to maintaining CFS reliability and minimizing challenges to reactor safety systems during power uprate operation. The NRC staff's review of the CFS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for power uprate operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 11.8 of the UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The CFS does not perform a safety-related function per se; however, its performance can have a major effect on plant availability and capability to operate reliably at the power uprate conditions, and failures in the CFS can result in loss of FW events and present challenges to reactor safety systems. The CFS is designed to provide sufficient FW at an elevated pressure and temperature to maintain the RV level within a predetermined range during all modes of power operation. The licensee has evaluated the capability of the CFS to perform its intended functions during power uprate operation based on a core power of 3952 MWt. Based on its evaluation the licensee found that it would be necessary to make some modifications to equipment in the CFS in order to attain the full EPU core thermal power. In order for the CFS to support the proposed EPU, the licensee has implemented the following modifications:

1. All three reactor feedwater pumps will be replaced in order to satisfy EPU feedwater demands. The instrumentation and control systems for these pumps and associated drive turbines will be re-calibrated and/or replaced in order to satisfy the flow requirements for EPU conditions. The turbine/pump coupling and the turbine diaphragms and buckets will also be replaced.

2. Two impellers in each of the three condensate pumps will be replaced. Also, each of the three condensate pumps will be fitted with new 1250 hp motors.

3. All three condensate booster pumps will be replaced. Also, each of the three condensate booster pumps will be fitted with 3000 hp motors.

With the above modifications, the licensee indicated that the CFS will meet the following criteria relative to the proposed power uprate:

1. The system will provide a reliable supply of feedwater at the increased reactor dome pressure, with sufficient capacity to accommodate steady-state flow requirements.

2. The CFS has sufficient capacity to provide the required feedwater flow.

3. The CFS is capable of providing adequate feedwater flow at the expected operating pressure, and sufficient margin exists so that a trip of one feedwater pump will not result in a reactor trip.

4. The runout capacity of the CFS in the limiting pump alignment will not exceed the performance capacity assumed in the transient analysis.

The licensee evaluated the CFS to ensure that a minimum of 5-percent margin above the required FW flow rate is available. The licensee also performed a transient analysis to confirm acceptable reactor water level response following a single FW pump trip.

Considering the extent of modifications that were being made to the CFS in support of EPU operation, the NRC staff requested that the licensee provide additional information concerning testing that will be performed to demonstrate acceptable transient response of the CFS at the uprated power level. In a letter dated September 27, 2006, the NRC staff informed the licensee that condensate, condensate booster, and reactor FW pump trip tests will be performed at 105-percent OLTP.

Based on a review of the information that was provided, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed power uprate on the capability and reliability of the CFS to provide reactor FW for uprated power operation. The modifications that are being made to the CFS are appropriate and necessary in order to maintain the capability and reliability of the CFS, and to minimize challenges to reactor safety systems. The NRC staff also finds that the transient testing that is planned to confirm appropriate CFS performance at the uprated power level is necessary due to the extent of modifications that are required. Because the NRC staff is relying in part on satisfactory completion of CFS transient testing in determining that the CFS is acceptable for uprated power operation, a License Condition will be established to require the satisfactory completion of the transient tests that are deemed to be necessary. This License Condition is described and further discussed in Section 2.12.

<u>Conclusion</u>

The NRC staff has reviewed the licensee's assessment of the effects of the proposed power uprate on the CFS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the CFS. The NRC staff concludes that the CFS will continue to maintain its ability to satisfy FW requirements for normal operation and shutdown, withstand water hammer, maintain isolation capability in order to preserve the system safety function, and not cause failure of safety-related SSCs. The NRC staff further concludes that the CFS will continue to meet the provisions of draft GDC-4, 40, and 42. Therefore, the NRC staff finds the CFS acceptable for operation at uprated conditions.

2.5.5 Waste Management Systems

2.5.5.1 Gaseous Waste Management Systems

The gaseous waste management systems (GWMSs) involve the gaseous radwaste system, which deals with the management of radioactive gases collected in the offgas system or the waste gas storage and decay tanks. In addition, it involves the management of the condenser air removal system, the gland seal exhaust and the mechanical vacuum pump operation exhaust, and the building ventilation system exhausts.

Regulatory Evaluation

The NRC staff's review focused on the effects that the proposed uprate may have on (1) the design criteria of the GWMSs; (2) methods of treatment; (3) expected releases; (4) principal parameters used in calculating the releases of radioactive materials in gaseous effluents; and (5) design features for precluding the possibility of an explosion if the potential for explosive mixtures exists. The NRC's acceptance criteria for GWMSs are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) draft GDC-3, insofar as it requires that the reactor facility shall be designed (1) to minimize the probability of events, such as fire and explosions; (2) to minimize the potential effects of such events to safety; (3) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; (4) draft GDC-67, 68, and 69, insofar as they require that systems that contain radioactivity be designed with appropriate confinement; and (5) 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion. Specific review criteria are contained in SRP Section 11.3.

Technical Evaluation

During normal operation, the GWMS collects and processes gaseous radioactive waste from the main condenser air ejectors, the startup vacuum pumps, condensate drain tank vent, and the steam packing exhauster, and control their release to the atmosphere through the plant stack so that the total radiation exposure to persons outside the controlled area is as low as reasonably achievable and the releases due to normal plant operation remain below the limits of 10 CFR Part 20, and 10 CFR Part 50, Appendix I. The proposed power uprate does not result in changes in operation or design of the GWMS. In Enclosure 1 to the March 7, 2006, letter, the licensee states that the offgas system is designed to control the release of plant-produced radioactive material within the release limits specified in the Offsite Dose Calculation Manual, which will ensure the 10 CFR Part 20 and 10 CFR Part 50, Appendix I, requirements are met.

The offgas system flow rate is not power dependent, but rather a function of fuel cladding performance, main condenser air inleakage, charcoal absorber inlet dew point, and charcoal absorber temperature. The main condenser inleakage paths are not affected by power uprate, since the internal condenser vacuum levels are not changed by power uprate operation. Because the condenser air inleakage and dynamic absorption coefficient do not change as a result of uprated operation, absorber holdup times are unaffected and thus, off-gas system operation is not affected by the proposed uprate. While the volume or flow rate of gaseous radwaste is not increased by power uprate, the activity of the gaseous effluents may increase by as much as the percentage increase in power. However, the release of gaseous effluents will be maintained within the existing site release limits in accordance with existing administrative controls.

The licensee found that power uprate will affect the flow rate of radiolytic hydrogen and oxygen to the offgas system. Consequently, the catalytic recombiner temperature and offgas condenser heat load are affected. The licensee has performed an analysis of the offgas system utilizing a higher decomposition rate that is more conservative than the BFN specific decomposition rate and found that hydrogen flow rates and concentrations remain within the design limits of the offgas system for the uprated plant. Based on its evaluation, the licensee concluded that the catalytic recombiner and offgas condenser, as well as the downstream components, have sufficient design margin to handle the increase in thermal power without exceeding the system design temperature and that the gaseous radwaste system will continue to satisfy the plant licensing basis.

Based on a review of the information that was submitted, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed power uprate on the capability of the GWMS to perform its functions. Because the increase in offsite dose will remain well within limits, hydrogen flow rates and concentrations will remain within the design capability of the GWMS, and radiological release rates will continue to be administratively controlled during uprated power operation, the NRC staff finds that the GWMS will continue to satisfy the plant licensing basis following implementation of the proposed power uprate.

Conclusion

The NRC staff has reviewed the licensee's assessment related to the GWMSs. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of gaseous waste on the abilities of the systems to control releases of radioactive materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. The NRC staff finds that the GWMSs will continue to meet their design functions following implementation of the proposed power uprate. The NRC staff further concludes that the licensee has demonstrated that the GWMSs will continue to meet the requirements of 10 CFR 20.1301; draft GDC-3, 67, 68, 69, and 70, and 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D. Therefore, the NRC staff finds the proposed uprate acceptable with respect to the GWMSs.

2.5.5.2 Liquid Waste Management Systems

Regulatory Evaluation

The liquid waste management system (LWMS) consists of process equipment and instrumentation necessary to collect, process, monitor, store, recycle, and/or dispose of liquid radioactive waste. Major components include floor and equipment drains, transfer pumps, and various waste system tanks. The NRC staff's review of the LWMS for proposed power uprates focuses on the effects that the proposed power uprate may have on previous analyses and considerations related to the processing and management of liquid radioactive wastes; such as expected releases and principal considerations used in estimating the increase in volume of the liquid radioactive waste that will be released. The criteria that are most applicable to the NRC staff's review of LWMS for proposed power uprates are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified limits; (2) 10 CFR Part 50, Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the "as low as reasonably achievable" criteria; (3) draft GDC-70, Control of Releases of Radioactivity to the Environment (Category B), insofar as it specifies that the plant design should include means to control the release of radioactive effluents; (4) draft GDC-67, Fuel and Waste Storage Decay Heat (Category B), draft GDC-68, Fuel and Waste Storage Radiation Shielding (Category B), and draft GDC-69, Protection Against Radioactivity Release from Spent Fuel and Waste Storage (Category B), insofar that they specify that systems that contain radioactivity be designed with appropriate confinement; and (5) other licensing-basis criteria that apply. The NRC staff's review of the LWMS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for power uprate operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 9.2 of the UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The LWMS is designed to collect, process, recycle and dispose of radioactive liquid waste in accordance with the requirements outlined in 10 CFR Part 20 and in 10 CFR Part 50, Appendix I, and in accordance with the criteria specified by draft GDC-70. The information that was provided indicated that the proposed power uprate will not change the operation or design of the equipment used in the LWMS, the radiological and environmental monitoring of the waste streams will not be affected, and no new or different radiological release paths will be introduced as a result of the proposed power uprate. However, the licensee determined that the proposed power uprate will cause the volume of liquid radioactive waste to increase due to more frequent backwashing of condensate demineralizers (the largest source of additional liquid radioactive waste), and more frequent backwashing of the RWCU filter-demineralizers.

In the June 28, 2004, submittal, the licensee indicated that the proposed power uprate will cause the volume of liquid processed waste to increase. However, the licensee stated that the volume of fluid flowing into the liquid radwaste system will not increase significantly as a result of the power uprate. Since the design and operation of the LWMS will not change, the licensee concluded that the capacity of the LWMS will continue to be adequate.

In Enclosure 1 of the March 7, 2006, letter, the licensee states that under current operational practices, liquid radwaste is released periodically from the plant under controlled conditions as a planned evolution. Such radioactive liquid effluents are controlled on a batch basis and each batch is sampled and analyzed prior to discharge. The limits for each release are defined to keep radioactive material concentrations in the discharge canal as low as practicable and below the limits given in 10 CFR Part 20.

Based on a review of the information that was submitted, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed power uprate on the capability of the LWMS to perform its functions. Because the increase in additional radioactive waste being generated due to power uprate operation is expected to be minimal and well within the capacity of the liquid radioactive waste processing system, any increase in offsite dose projections as a consequence is expected to be inconsequential and remain well below established plant release limits.

Conclusion

The NRC staff has reviewed the licensee's assessment related to the LWMS. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of liquid waste on the ability of the LWMS to control releases of radioactive materials. The NRC staff finds that the LWMSs will continue to meet their design functions following implementation of the proposed uprate. The NRC staff further concludes that the licensee has demonstrated that the LWMS will continue to meet the requirements of 10 CFR 20.1301, draft GDC-67, 68, 69, and 70; and 10 CFR Part 50, Appendix I, Sections II.A and II.D. Therefore, the NRC staff finds the LWMSs acceptable for uprated conditions.

2.5.5.3 Solid Waste Management Systems

Solid radioactive waste consists of wet and dry waste. Wet waste consists mostly of low specific activity spent secondary and primary resins and filters, and oil and sludge from various contaminated systems. The NRC staff review relates primarily to wet waste dewatering and liquid collection processes, and focuses on the impact that the proposed power uprate will have on the release of radioactive materials to the environment via gaseous and liquid effluents. Because Sections 2.5.5.1 and 2.5.5.2 fully encompass these considerations, a separate evaluation of solid waste management systems in this section is not required.

2.5.6 Additional Considerations

2.5.6.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Nuclear power plants are required to have redundant onsite emergency power supplies of sufficient capacity to perform their safety functions (e.g., diesel engine-driven generator sets). The NRC staff's review for proposed power uprates focuses on the effects that the proposed power uprate may have on the fuel oil storage requirements for the EDGs. The licensee indicated that the fuel oil consumption rate is based on the electrical rating of the EDG and, because the electrical rating of the EDG is not affected by the proposed power uprate, the fuel oil consumption rate of the EDGs is not affected. Consequently, the existing fuel oil storage requirements are also not affected. Therefore, an evaluation of the EDG fuel oil storage requirements for the proposed power uprate is not required.

2.5.6.2 Light Load Handling System (Related to Refueling)

The light load handling system includes components and equipment used for handling new fuel at the receiving station and for loading spent fuel into shipping casks. Because the licensee is not introducing any new fuel designs in conjunction with the proposed power uprate, this area of review is not affected by the proposed power uprate and an evaluation of the light load handling system is not required.

2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The NRC staff's review for the primary containment functional design covered (1) the temperature and pressure conditions in the drywell and wetwell due to a spectrum of postulated LOCAs; (2) the differential pressure across the operating deck for a spectrum of LOCAs (Mark II containments only); (3) suppression pool dynamic effects during an LOCA or following the actuation of one or more RCS safety/relief valves; (4) the consequences of an LOCA occurring within the containment (wetwell); (5) the capability of the containment to withstand the effects of steam bypassing the suppression pool; (6) the suppression pool temperature limit during RCS safety/relief valve operation; and (7) the analytical models used for containment analysis. The NRC's acceptance criteria for the primary containment functional design are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of an LOCA; (2) draft GDC-10, insofar as it requires that reactor containment be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other ESFs as may be necessary, to retain for as long as the situation requires the functional capability; (3) draft GDC-49, insofar as it requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following an LOCA, including considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of ECCSs; (4) draft GDC-12, insofar as it requires that instrumentation and controls be provided as required to monitor and maintain variables within prescribed operating ranges; and (5) draft GDC-17, insofar as it requires that means be provided to monitor the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents. Specific review criteria are contained in SRP Section 6.2.1.1.C.

Technical Evaluation

The primary containment is a Mark I design consisting of (1) a drywell which encloses the reactor vessel, the reactor coolant system and other branch connections to the reactor coolant system; (2) a toroid-shaped pressure suppression chamber (or wetwell) partially filled with a large volume of water (the suppression pool) and a toroid-shaped ECCS ring header circumscribing the suppression chamber which is the primary source of water for the ECCS low head pumps; (3) a vent system connecting the drywell atmosphere to the suppression chamber; (4) containment isolation valves; (5) containment cooling systems; and (6) other equipment.

The proposal to operate at power uprate conditions requires that safety analyses for those DBAs whose results depend on power level be recalculated at the higher power level. The containment design basis is primarily established based on the LOCA and the actuation of the reactor vessel safety relief valves (SRVs) and their discharge into the suppression pool.

Short-term and long-term containment analyses results are reported in the UFSAR. The short-term analysis is directed primarily at determining the drywell pressure response during the initial blowdown of the reactor vessel inventory to the containment following a large break inside the drywell. The long-term analysis is directed primarily at the suppression pool temperature response, considering the decay heat addition to the suppression pool. The effect of power uprate on the events yielding the limiting containment pressure and temperature responses are provided below.

Short-term LOCA Analysis

The short-term LOCA analysis is performed for the limiting DBA LOCA, which assumes a double-ended guillotine break of a recirculation suction line, to show that the peak drywell pressure and temperature remain below the drywell design pressure of 56 psig and the drywell design temperature of 281 degrees F. 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 2-percent above EPU rated thermal power (RTP), using analytic methods approved for EPUs and a 30 psig increase in operating steam dome pressure. The licensee used the LAMB computer code for the short-term mass and energy release and the M3CPT computer code for the containment response. The power uprate methods approved by the NRC permit the use of either the M3CPT computer code or the LAMB computer code to calculate the mass and energy release from the postulated pipe break into the drywell. The licensee has used the Moody slip critical flow model. The Moody slip critical flow model is conservative compared to more realistic prediction methods such as the homogeneous equilibrium model. The homogeneous equilibrium model was used for break flow calculations as part of the Mark I Long-Term Program to address containment hydrodynamic loads.

The results of these analyses at EPU and the acceptance criteria are provided in Table 4-1 of the PUSAR. The short-term portion of this table is reproduced below.

BROWNS FERRY SHORT-TERM LOCA CONTAINMENT PERFORMANCE RESULTS

Parameter	OLTP [(1)] (Historical)	EPU (Current Methods)	Design Limit
Peak Drywell Pressure (psig)	49.6	48.5[(3)]	56
Peak Drywell Air Space Temperature (degree F)	294	295.2[(3)]	340/281[(2)]

(1) Unit 1 UFSAR Section 14.11.3 values.
(2) The acceptance limit for drywell air space temperature is 340 degrees F, while the shell design acceptance limit is 281 degrees F. The listed peak values are for air space temperature.
(3) LAMB mass and energy release data used as input to M3CPT.

The table compares the peak pressure and temperature at the OLTP and using acceptable current calculation methods at EPU. The results of these calculations show that the peak drywell pressure at power uprate conditions remains below the respective design limits. The drywell air space temperature exceeds the structural design temperature for less than one minute. The licensee states that this is insufficient time for the drywell structure to reach the 281 degrees F design limit.

The use of the reactor and containment conditions corresponding to the EPU is conservative with respect to the proposed 5-percent power uprate since the higher power results in higher temperatures and pressures than those expected with a 5-percent power uprate.

Based on the use of acceptable calculation methods and conservative assumptions, and results less than the design containment pressure and temperature, the short-term containment response at 5-percent power uprate is acceptable.

Long-term LOCA Analysis

The long-term LOCA analysis was performed for the DBA LOCA at 2 percent above the EPU RTP. The SHEX computer code is used for the analysis of the peak suppression pool temperature, long-term peak wetwell pressure and peak wetwell air temperature. The NRC has accepted this computer code for previous power uprate applications. The licensee used the American National Standards Institute/American Nuclear Society (ANSI/ANS) 5.1-1979 decay heat model with a 2σ uncertainty added. The licensee incorporated the guidance of Service Information Letter 636 Revision 1, which recommends accounting for additional actinides and activation products that further increases the predicted decay heat.

The long-term LOCA analysis demonstrates that the peak suppression pool temperature, and wetwell pressure remain below their respective design limits. The results of these analyses and the acceptance criteria are provided in Table 4 -1 of the PUSAR. The relevant portions of this table are reproduced below.

CONTAINMENT PERFORMANCE RESULTS

Parameter	OLTP	EPU	Design Limit
Peak Bulk Pool Temperature for Design Basis LOCA (degree F)	170.0	187.3 (4), (5)	281
Long-term Peak Wetwell Pressure for Design Basis LOCA (psig)	27	30.5	56

(4) Calculation uses the ANSI/ANS 5.1 decay heat model (with a 2σ uncertainty).
(5) A service water temperature of 95 degrees F was used.

The wetwell pressure peaks early in the event, and then peaks again around the time at which the wetwell temperature peaks. The value of the second peak is the highest long-term wetwell pressure and is presented in the table.

The EPU peak suppression pool temperature of 187.3 degrees F is less than the torus design temperature of 281 degrees F. The long-term wetwell air space temperature will also be 187.3 degrees F since thermal equilibrium is conservatively assumed between the wetwell air space temperature and the suppression pool temperature. The wetwell air space temperature is therefore less than the design limit of 281 degrees F. The secondary (long-term) wetwell air space peak pressure is 30.5 psig, which is well below the torus design pressure limit of 56 psig.

The most limiting drywell air space temperature is a result of small steam line breaks. The peak drywell air space temperature for these breaks is 336 degrees F which occurs prior to containment spray initiation. The resultant peak shell temperature calculated is 277.1 degrees F which is below the design value of 281 degrees F. The drywell air space temperature is used to assess the EQ of equipment. It should be noted that the peak drywell air temperature given in the previous tables was for the DBA break which is the double-ended guillotine break of a recirculation suction line.

The use of the reactor and containment conditions corresponding to the EPU is conservative with respect to the proposed 5-percent power uprate since the higher power results in higher temperatures and pressures.

Since the licensee used acceptable calculation methods and conservative assumptions and the calculated values are below the design limits, the long-term containment calculations for power uprate conditions are acceptable.

<u>Hydrodynamic Loads</u>

Part of the containment design basis is the acceptable response of the containment to hydrodynamic loads associated with the discharge of reactor steam and drywell nitrogen into the suppression pool following an LOCA or the discharge of reactor steam following actuation of the SRVs. Analytical and empirical methods, approved by the NRC staff in NUREG-0661 were used by the licensee to address these issues for Unit 1 and to develop a plant unique structural evaluation. The NRC staff found the resolution of these issues to be acceptable.

The licensee, as part of the power uprate evaluation, must ensure that these analyses remain bounding. This is done for the LOCA by means of short-term calculations of the pressure and temperature response to a double-ended break of a reactor coolant system recirculation line. The key parameters are the drywell and wetwell pressure, vent flow rates and the suppression pool temperature. The pressures are influenced by the 30-psi increase in dome pressure.

In section 4.1.2.1 of the PUSAR, the licensee states that:

> . . . the short-term DBA-LOCA containment responses for EPU are within the range of test conditions used to define the pool swell and CO [condensation oscillation] loads for Browns Ferry. The containment responses with EPU, in which chugging would occur, are within the conditions used to define chugging loads. The vent thrust loads with EPU are calculated to be less than plant-specific values defined for Browns Ferry.

The use of the reactor and containment conditions corresponding to the EPU is conservative with respect to the proposed 5-percent power uprate since the higher power results in higher temperatures and pressures than those expected with a 5-percent power uprate. The licensee's evaluation of containment hydrodynamic loads as a result of an LOCA are consistent with ELTR1 and ELTR2, show acceptable results, and are therefore acceptable for the power uprate.

Main Steam Relief Valve Loads

The dynamic loads on the suppression pool due to the discharge of steam from MSRVs are part of the containment design basis. The SRV loads are evaluated for two cases: initial actuation and re-actuation. An increase in the MSRV opening setpoint pressure results in higher MSRV flow rates, and therefore, higher MSRV loads. For Unit 1, the licensee increased the SRV opening setpoint pressures by 30 psi. The licensee indicated that the increased MSRV loads resulting from this increase in the setpoint pressures were compared with plant unique design limits calculated during the Mark I Containment Long-Term Torus Integrity Program. The comparison shows there is sufficient conservatism in the OLTP containment MSRV load definition to accommodate the increased MSRV loads due to power uprate. Therefore, power uprate does not affect the first actuation MSRV load definitions. Since the reactor pressure and MSRV setpoints are the same for power uprate, the same conclusion applies.

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 EPU is the time between MSRV actuations. A higher water level at the time of second pop will result in higher MSRV loads. The licensee stated that the effect of the power uprate on the MSRV discharge line was conservatively evaluated. The increased MSRV loads resulting from subsequent actuations were compared with plant unique design limits calculated during the Mark I Long-Term Torus Integrity Program. The comparison also shows there is sufficient conservatism in the OLTP containment MSRV load definition to accommodate the increased MSRV loads due to subsequent actuations. Therefore, the existing load definition for SRV initial actuation and re-actuation remain applicable.

Local Pool Temperature with MSRV Discharge

A local pool temperature limit for MSRV discharge is specified in NUREG-0783 because of concerns resulting from unstable condensation observed at high pool temperatures in BWRs without quenchers. The licensee indicated that the peak local suppression pool temperature at Unit 1 has been evaluated for power uprate and meets the NUREG-0783 criteria. The MSRV flow capacities and the configuration of the T-quenchers remain unchanged for power uprate and the predicted local pool temperatures remain within the 200 degrees F limitation. Therefore, the peak local suppression pool temperature at Unit 1 is acceptable for the power uprate conditions.

It is necessary to ensure that steam ingestion in the ECCS suction line is not of concern during MSRV steam discharge at high suppression pool temperature because the tops of the ECCS suction strainers at Unit 1 are located above the T-quenchers. The licensee evaluated the configuration of the suppression pool, MSRV T-quenchers, and ECCS suction strainers utilizing information contained in TRs NEDO-30832, *Elimination of Limit on Suppression Pool Temperature for SRV Discharge with Quenchers*, and NEDO-31695, *BWR Suppression Pool Temperature Technical Specification Limits*, which were approved by the NRC in a letter dated August 29, 1994. Based on this evaluation, the licensee concluded that the ECCS suction piping would not ingest steam bubbles that could later collapse and induce water hammer loads. The flow rates from the T-quenchers and the configuration of the T-quenchers in relation to the suction strainers is the same as that for Units 2 and 3.

The analyses done by the licensee have resulted in a higher CS flow in the short-term. However, the NRC staff judges that this increase is not significant in terms of the approach velocity to the strainers when considering steam ingestion. Therefore, the conditions do not affect the criterion used to address steam ingestion into the suction strainers, and the conclusions remain valid for the power uprate conditions.

The licensee has not proposed any changes to instrumentation and controls provided to monitor and maintain variables within prescribed operating ranges. The licensee has also not proposed any changes to instrumentation provided to monitor the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment temperature and pressure transient and concludes that the licensee has adequately accounted for the increase of mass and energy resulting from the proposed power uprate. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and the containment and associated systems will continue to meet the provisions of draft GDC-10, 12, 17, 40, 42, and 49 following implementation of the proposed power uprate. Therefore, the NRC staff finds the primary containment functional design acceptable for power uprate conditions.

2.6.2 Subcompartment Analyses

Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The NRC staff's review for subcompartment analyses covered the determination of the design differential pressure values for containment subcompartments. The NRC staff's review focused on the effects of the increase in mass and energy release into the containment due to operation at power uprate conditions, and the resulting increase in pressurization. The NRC's acceptance criteria for subcompartment analyses are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of an LOCA; and (2) draft GDC-49, insofar as it requires that the containment structure, including access openings and penetrations, and any necessary containment heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following an LOCA. Specific review criteria are contained in SRP Section 6.2.1.2.

Technical Evaluation

An annular structure of reinforced concrete enclosed in steel plate inside the drywell, called a sacrificial or biological shield, provides thermal and radiation shielding. Section 12.2.2.6 of the UFSAR describes the sacrificial shield as well as an analysis of the capability of the sacrificial shield to withstand the differential pressure which would develop across the wall as a result of a high pressure pipe break between the reactor vessel and the shield wall. This differential pressure is a function of the break size and the annular vent area to the rest of the drywell.

The UFSAR states the effects of postulated LOCAs occurring within the sacrificial shield area have been investigated. The only safe-end-to-nozzle welds, safe ends, or piping located in the annulus are small diameter lines whose rupture would result in relatively small pressure differences. The largest line which has the safe end located in the annulus is the 4-inch jet pump instrument line nozzle. For all larger lines, the double-ended line break results in the flow being directed into the drywell and not into the annulus.

The licensee evaluated the annulus pressure load on the biological shield wall due to a postulated break in a 4-inch jet pump instrument line nozzle at EPU conditions. The annulus pressure load (2.4 pounds per square inch differential (psid)) evaluated in UFSAR Section 12.2.2.6 remains bounding compared to the annulus pressure load of 2.3 psid for normal FW temperature at 2-percent above the EPU power. For final FW temperature reduction (FFWTR), the annulus pressure load is 2.6 psid at 2 percent above the EPU power. Higher results for EPU are due to the additional conservatism of the FFTWR input values. The licensee used slightly higher subcooling which resulted in a higher critical mass flux. This is an additional evaluation which was not previously performed. The biological shield and component design loads at EPU conditions remain well below the Unit 1 design basis value of 19 psid. These conditions and calculation results are conservative with respect to those at power uprate.

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment temperature and pressure transient and concludes that the licensee has adequately accounted for the increase of mass and energy resulting from the proposed power uprate. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and the containment and associated systems will continue to meet the provisions of draft GDC-10, 12, 17, 40, 42, and 49 following implementation of the proposed power uprate. Therefore, the NRC staff finds the primary containment functional design acceptable to support power uprate operation.

2.6.3 Mass and Energy Release

Mass and Energy Release Analysis for Postulated Loss of Coolant

The licensee's evaluation of postulated pipe failures was based on a power level of 3952 MWt. The uprated plant requires a small (less than 3 percent) increase in RPV dome pressure to supply sufficient steam to the main turbine for operating at power uprate conditions. The slight increase in vessel pressure and temperature will result in a small increase in the mass and energy release rates following postulated pipe failures. The licensee's evaluation of the impact that power uprate will have on the consequences of high and moderate energy piping failures located outside containment is discussed in Sections 10.1 and 10.2 of the PUSAR. In a letter dated July 26, 2006, the licensee indicated that no new break locations in the MS and FW piping are required to be postulated due to the proposed power uprate and that plant walk-downs will be performed to confirm that pipe whip restraints have been installed with no significant changes in configuration from what was specified on the original drawings.

The licensee's evaluation of internal flooding due to high energy line breaks is addressed in Section 10.1.3 of the PUSAR. The licensee determined that the RWCU and the reactor FW systems are the only two high energy systems with liquid filled lines that pose a flooding concern. In a letter dated February 23, 2005, the licensee indicated that the proposed power uprate will not result in any significant changes (less than 1-inch) in internal flooding levels and mechanical equipment will not be prevented from performing their necessary safety-related functions. The licensee's conclusion is based on evaluation of both high and moderate energy pipe failures.

Based on a review of the information that was provided, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed power uprate on the consequences of postulated high and moderate energy pipe failures, including flooding considerations. The licensee determined that the proposed power uprate will not result in any new pipe failure locations, and the consequences of postulated pipe failures will not exceed plant design limitations that were previously recognized and credited.

2.6.4 Combustible Gas Control in Containment

Regulatory Evaluation

Following an LOCA, hydrogen and oxygen may accumulate inside the containment due to chemical reactions between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excessive hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The NRC staff's review covered (1) the production and accumulation of combustible gases; (2) the capability to prevent high concentrations of combustible gases in local areas; (3) the capability to monitor combustible gas concentrations; and (4) the capability to reduce combustible gas concentrations. The NRC staff's review primarily focused on any impact that the proposed power uprate may have on hydrogen release assumptions, and how increases in hydrogen release are mitigated. The NRC's acceptance criteria for combustible gas control in containment are based on (1) 10 CFR 50.44, insofar as it requires that plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere; and (2) Specific review criteria are contained in SRP Section 6.2.5.

Technical Evaluation

The post-LOCA production of hydrogen and oxygen by radiolysis increases proportionally with the power level. The hydrogen concentration in containment is controlled by the Containment Atmosphere Dilution (CAD) system which is described in Section 5.2.6 of the FSAR. Because of the increased production of hydrogen and oxygen due to the power uprate the system must be started sooner after the beginning of the accident. This does not significantly affect operator response since the system is not required for many hours after accident initiation.

The licensee analyzed the post-LOCA control of combustible gases at EPU conditions. The use of the reactor and containment conditions corresponding to EPU is conservative with respect to the proposed power uprate since the higher power results in the generation of more hydrogen. The results of the combustible gas analyses are given in Section 4.7 and in Figures 4-1 through 4-4 of the PUSAR. For Unit 1, the required start time of the CAD system following an LOCA decreases from 42 hours to 32 hours, which is adequate time for effective operator response. More time to respond would be available at power uprate conditions. Per UFSAR Sections 5.2.6.1.c and 5.2.6.1.g, the design basis for the CAD system requires that the system be designed for a possible startup 10 hours after an LOCA, and that containment pressure shall not exceed 30 psig as a result of CAD system operation. The licensee indicated that EOI, Appendix-14 B (CAD Operation), contains measures to ensure CAD system operation does not result in containment pressure exceeding 30 psig. For DBA LOCA conditions, which are bounding for containment pressure, the drywell reaches the 30 psig limit in 15 days at EPU with nitrogen addition without venting, compared to 18 days before EPU. This reduction in time does not affect any design basis requirements and is therefore acceptable.

Conclusion

The NRC staff has reviewed the licensee's assessment related to combustible gas and concludes that the plant will continue to have sufficient capabilities consistent with the requirements in 10 CFR 50.44 and 10 CFR 50.46 for systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure containment integrity is maintained at power uprate.

2.6.5 Containment Heat Removal

Regulatory Evaluation

RHR systems are provided to remove heat from the containment atmosphere and from the water in the containment wetwell. The NRC staff's review in this area focused on the effects of the proposed power uprate on the analyses of the available NPSH to the RHR and containment spray system pumps. The NRC's acceptance criteria for containment heat removal at Browns Ferry are based on draft GDC-10, 49 and 52, insofar as they require that a containment heat removal system be provided, and that its function shall be to prevent exceeding containment design pressure under accident conditions. Specific review criteria are contained in RG 1.82 Revision 3. The licensee has not adopted RG 1.82 Revision 3, however, the licensee has addressed the degree of conformity with the guidance of this regulatory guide in a letter to the NRC and concluded that the request for EPU complies with this regulatory guide's positions on NPSH for the RHR and CS pumps. This evaluation bounds power uprate conditions.

Due to the Unit 1 extended shutdown, the licensee had not responded to several NRC generic communications related to the NPSH of ECCS and containment heat removal pumps on the original schedules for these generic communications, in particular, NRC BL 93-02, BL 96-03, and GL 97-04. The licensee subsequently responded to BL 93-02 for Unit 1 by a letter to the NRC dated May 6, 2004. The NRC concluded that the licensee had acceptably resolved the issues raised by BL 93-02 in a July 26, 2006, letter to TVA. The NRC staff addresses BL 96-03 and GL 97-04 later in this section.

<u>Technical Evaluation</u>

RHR and Core Spray System Description

In a letter dated March 23, 2006, the licensee provided a summary description of the containment and ECCS designs. A more detailed description is provided in the UFSAR.

The water source for each unit's ECCS is a ring header circumscribing the suppression chamber with connecting piping to four inlet penetrations through the torus wall into the suppression pool. Inside the suppression pool, each of these four connecting lines is fitted with an ECCS suction strainer at a flanged joint. The four strainers are not associated with individual pumps. The ECCS ring header supplies suppression pool water to the suction piping of the RHR, CS, HPCI and RCIC systems.

The RHR system is described in Section 4.8 of the UFSAR. The RHR system consists of two trains. Each train consists of two parallel flow paths, each with an RHR pump and heat exchanger. Both heat exchangers in an RHR train are cooled by one train of the RHR service water system. The RHR systems of the three BFN units are cross-tied for additional redundancy.

The RHR system has several different modes of operation. It is aligned during normal operation as part of the ECCS in the low-pressure coolant injection (LPCI) mode. The RHR pumps are sized for this function. The RHR system also cools the reactor coolant system during a normal shutdown and cooldown. In addition, it cools the suppression pool by pumping suppression pool water through the RHR heat exchangers and returning the water to the suppression pool, or by diverting the suppression pool water to spray headers in the drywell and wetwell after it passes through the RHR heat exchangers. The RHR heat exchangers are sized based on cooling the suppression pool following a design basis LOCA.

The Unit 1 RHR pumps are equipped with discharge flow limiting orifice plates to prevent pump runout by limiting the flow into a broken recirculation loop during an LOCA.

The low pressure CS system consists of two independent and redundant trains. Each train contains two 50-percent capacity CS pumps, a spray header or sparger inside the core shroud above the core and piping and valves to transport water from the suppression pool to these spray headers. The water returns to the suppression pool by flowing out the break. The CS system is not equipped with orifices.

Available Net Positive Suction Head (NPSH)

An important consideration in the operation of the CS and RHR pumps is the available NPSH. Adequate available NPSH is important in ensuring that the pump will deliver the flow rate assumed at the expected discharge pressure. In order to ensure acceptable flow and discharge pressure, the available NPSH should be equal to or greater than the required NPSH. The available NPSH is a function of the system design and operation. The required NPSH is a function of the pump design and is determined by tests performed by the pump vendor. The required NPSH increases as the flow rate increases.

The available NPSH is calculated from the equation

$$\text{Available NPSH} = h_{atm} + h_{static} - h_{loss} - h_{vapor}$$

where

h_{atm} = the head on the surface of the suppression pool due to the pressure in the wetwell atmosphere

h_{static} = the head due to the difference in elevation between the suppression pool surface and the centerline of the pump suction

h_{loss} = the head loss due to fluid friction, fittings in the flow path from the suppression pool to the pump, and the ECCS suction strainers which prevent ingestion of debris into the pumps

h_{vapor} = the head due to the vapor pressure of the suppression pool water at the suppression pool water temperature

All head values are measured in feet of water.

Required NPSH

Industry standards for determining required NPSH typically cite a value corresponding to a 3-percent loss in total head. The licensee has selected values of required NPSH less than this value (i.e., corresponding to a head loss greater than 3 percent).

The required NPSH values for the RHR and CS pumps are based on curves provided by the pump vendor. The required NPSH determined by the pump vendor is a function of the operating time and the flow rate. The curves are provided in Enclosure 1, TVA Calculation MDQ099920060011, of the August 4, 2006, letter. The curves are taken from the pump vendor's report assessing the capability of the RHR and CS pumps to operate for up to 8000 hours (approximately 1 year) at the proposed required NPSH values.

The licensee divides the LOCA analyses into a short-term period and a long-term period. The short-term is the time from the initiation of the LOCA to 10 minutes. During this time the ECCS starts automatically upon receiving an initiating signal and operates with no operator intervention. After 10 minutes, during the long-term portion, the operator is assumed to adjust flow rates and to reconfigure the RHR system from an injection mode to the containment spray mode which also cools the suppression pool. Limiting NPSH conditions may occur during either the short-term or long-term portions of the LOCA.

The licensee has determined that, for the short-term portion of the LOCA, the RHR pumps pumping to the broken loop do not have adequate available NPSH even when credit is taken for the pump vendor reduced required NPSH values and for containment accident pressure. In this case, the licensee also credits the pump vendor's evaluation which concludes that the pumps will be capable of performing their long-term safety function following a brief period at significantly reduced available NPSH (i.e., the pumps will be cavitating). In addition, the licensee presented data from TVA tests which provide additional confidence that the pumps will be capable of performing their long-term safety function following a brief period of significantly reduced available NPSH. These tests are described in TVA reports provided to the NRC.

The TVA tests used Unit 3 RHR pump 3A. The purpose of the tests was to show that the pump could withstand operation with cavitation below the 3-percent head drop level during the short-term LOCA period, and subsequently perform its safety function during the long-term LOCA period after the operator has reduced the pump flow rate. This is the same situation which would exist given the power uprate.

TVA performed the tests *in situ* in Unit 3. The testing consisted of two parts. In Part A, the licensee attempted to duplicate the pump's total dynamic head (TDH) curve obtained by the pump vendor which plots pump total dynamic head against pump flow rate. In Part B, the licensee operated the pump at several levels of cavitation. For both tests the pump was operated in the test return mode. In this mode of operation, the pump takes suction from the suppression pool and returns the flow to the suppression pool via a full flow test line.

The Part A tests successfully reproduced the TDH curve obtained by the pump vendor as shown in Figure 5.1 of the licensee's July 21, 1976, report. The Part B tests were constant flow rate tests at 8,000 gallons per minute (gpm) and 10,000 gpm in which the pump suction pressure was adjusted (and cavitation conditions were obtained) by closing a suction gate valve. Table 5.2 of the licensee's July 21, 1976, report provides the results of the licensee's testing. RHR pump motor vibration during the tests was measured. The degree of cavitation could be judged qualitatively by the measured suction pressure, level of vibration and by emitted sound. The Part B test report stated that

> . . . in all cases the pump motor vibration displacements and accelerations do not exceed the GE recommended vibration acceptance criteria for long-term pump motor reliability.

In a letter dated September 15, 2006, the licensee described the margin available to the RHR pumps during the short-term portion of the LOCA. This information, as well as information in the July 21, 1976, test report and the licensee's calculations is summarized in the following table.

SUMMARY OF NPSH MARGIN CONSIDERING TVA NPSH TESTS

(1) Vendor certified required NPSH (Table 5.1 of July 21, 1976, report)	(2) Required NPSH test value (obtained by extrapolation of test data -Table 3 of May 21, 1976, report)	(3) Required NPSH in short-term LOCA (Taken from Sulzer time dependent required NPSH study and extrapolated to maximum flow rate)	(4) Available NPSH (Value given to Sulzer by TVA for Sulzer time-dependent required NPSH study)	(1) - (2)	(4) - (2)	(4) - (3)
34.0 ft (@ 12,000 gpm)	25.0 ft (@ 12,000 gpm)	28.4 ft (@ 11,500 gpm)	26.4 ft (@ 11,500 gpm)	9.0 ft	+ 1.4 ft	- 2.0 ft

The required NPSH corresponding to a 3-percent head drop from the pump vendor's data is 34.0 ft at 12,000 gpm. The lowest value of required NPSH for RHR pump 3A measured in the TVA tests (adjusted to 12,000 gpm) is 25.0 ft. Thus, the TVA tests show a 9-foot margin below the 3-percent head drop value. These values are based on a flow rate of 12,000 gpm. The assumed flow rate of the RHR pumps during the short-term period of the LOCA is 11,500 gpm. The use of required NPSH values at the different flow rates for this comparison is acceptable since at 12,000 gpm the required NPSH would be greater than the required NPSH at 11,500 gpm. This underestimates the NPSH margin and is therefore conservative.

The licensee calculated that the available NPSH at 10 minutes, including credit for containment accident pressure is 26.4 ft at a flow rate of 11,500 gpm. The licensee extrapolated the pump vendor data to obtain the required NPSH at 10 minutes for a flow rate of 11,500 gpm. This required NPSH value is 28.4 ft. Thus, there is a negative margin of two ft at the end of the short-term LOCA period at 10 minutes (26.4 ft - 28.4 ft), causing the pump to cavitate.

To compensate for this negative margin, the licensee depends on the technical judgment of the pump vendor, who states that:

- Although vibration and noise should increase due to surging and cavitation from the transient event [the short-term LOCA], the units should continue operation;

- Some detrimental damage is likely, due to the transient event, but should not be catastrophic. After 10 minutes, if the operational life graph [the pump vendor's required NPSH curve] is followed the pumps will continue to function.

The pump vendor's opinion is reinforced by the 1976 TVA test data. Using these data, the available NPSH is greater than the required NPSH by 1.4 ft. Pump 3A was tested in cavitation at the lowest pump suction pressure for 10 minutes. The calculated time during the short-term LOCA period that the RHR pumps pumping to the broken loop would cavitate is approximately 4 minutes.

The NRC staff determined that the TVA test reports were acceptable in that the tests were carefully run, and the results appear reasonable and consistent with information on cavitation tests on similar pumps. In addition, the NRC staff requested TVA to address whether the 3A pump experienced any abnormal operation since this testing. The licensee performed a search of completed surveillances and work orders associated with pump 3A during the two years following the reduced NPSH testing and found no anomalies in surveillance testing or maintenance.

In a letter dated October 5, 2006, the licensee stated that:

> . . . in August/September 1994, the 3A RHR pump impeller was replaced to address wear ring cracking concerns. [A generic issue which was not specific to Browns Ferry.] A review of documentation associated with this replacement did not indicate any abnormal impeller wear.

The licensee also demonstrated with a sensitivity analysis that the deficit in available NPSH for the RHR pumps pumping into the broken loop during the short-term LOCA is a result of the conservatism in the analysis. The conservatism includes assuming:

- The reactor power (the EPU power level was used) is 2-percent greater than the RTP to account for instrument uncertainty;

- The decay heat is 2σ greater than the nominal value to account for uncertainty;

- The mixing of the broken loop flow with the drywell atmosphere is 100-percent efficient. This decreases the calculated containment accident pressure;

- The initial suppression pool temperature is at the TS limit: 95 degrees F;

- The suppression pool level corresponds to the minimum water volume specified in the TS;

- The assumed RHR pump flow for the NPSH analyses is 11,500 gpm. The calculated maximum flow for the broken loop RHR pumps is 11,000 gpm. A lower flow results in a lower required NPSH;

- The drywell relative humidity is 100 percent. This minimizes the amount of noncondensible gas in the drywell which reduces the initial drywell pressure.

Assuming that all these conditions occur simultaneously is additional conservatism.

The licensee performed a sensitivity study in which the last two conservative assumptions were modified. A pump flow rate of 11,000 gpm was assumed along with a 50-percent drywell relative humidity (the 50-percent relative humidity value is used by the licensee in the special events NPSH analyses discussed below). The result of this sensitivity study showed that sufficient containment pressure is available to provide positive NPSH margin without reliance on pump cavitation testing, and demonstrate the effect of the conservatism in the calculations.

Based on the above discussion, the NRC staff finds that taking into account the pump vendor's judgment as well as the TVA cavitation test results, the performance history of pump 3A after the cavitation tests, and the licensee's observations of the pump 3A impeller while replacing wear rings, the licensee's determination of NPSH margin for the short-term LOCA period is acceptable.

Analysis Methods

In order to calculate the available NPSH for a specific scenario, the containment conditions must first be determined (i.e., the drywell and wetwell pressure and suppression pool temperature). As shown in Table 1-3 of the PUSAR, the licensee calculates the containment conditions for the long-term portion of the LOCA with the GE SHEX computer code. The containment conditions for the ATWS, SBO and the Appendix R Fire are also calculated with SHEX.

Previously, the NRC staff performed an audit calculation of SHEX predictions of containment accident conditions for NPSH analyses as part of the review of another BWR's EPU using the NRC computer code CONTAIN 2.0. The results of the NRC staff's review show close agreement between the SHEX calculations and those done with CONTAIN 2.0.

As discussed in Enclosure 3 of the letter dated August 18, 2006, the licensee uses the MULTIFLOW computer program to calculate the flow losses for NPSH calculations. This is a TVA computer program which solves steady state hydraulic flow networks. The description of MULTIFLOW provided by the licensee states that it produces quality assured solutions for raw water and condensate systems. The NPSH analyses are for similar systems.

The licensee determined flow resistance in the piping system using generally recognized methods. The NRC staff questioned the licensee's use of a piping roughness of 0.00015 ft since this value corresponds to clean commercial steel. The licensee stated that this piping roughness value is acceptable for a condensate quality system and would not be expected to change with system age. The NRC staff questioned the applicability of this value to the suppression pool water. The licensee clarified that:

> . . . the suppression pool water, including any inventory added from the RCS [reactor coolant system] or the CST [condensate storage tank], is clean demineralized water.

In this case, the NRC staff finds that the licensee's roughness value is appropriate.

In a letter dated March 7, 2006, the licensee verified that input parameters not affected by the power uprate remain the same as those in the UFSAR. These include such things as containment volumes, heat sink modeling, heat exchanger effectiveness, etc. This results in a more conservative value for heat exchanger effectiveness since the heat exchanger effectiveness increases as the suppression pool water temperature increases.

Ultimate Heat Sink

The service water and UHS temperature limit for Unit 1 is specified in TS SR 3.7.2.1 as 95 degrees F. For the special events, the licensee assumes the historically highest service water temperature of 92 degrees F. The following table, taken from information in TVA NPSH Calculation MDQ099920060011 summarizes the UHS temperature assumed for each event for which the NPSH margin was analyzed for the RHR and CS pumps.

Loss-of-Coolant Accident (LOCA)

The increase in reactor power as a result of the power uprate results in an increase in the suppression pool temperature following the design basis LOCA. The increased water temperature reduces the available NPSH of the RHR pumps and the CS pumps since the vapor pressure of the suppression pool water (or h_{vapor}) increases with the increased suppression pool temperature. The licensee has proposed to compensate for this reduction in available NPSH by crediting the containment accident pressure (which increases h_{atm}) when necessary to ensure that the available NPSH is equal to the required NPSH. In discussing credit for containment accident pressure, the licensee uses the term overpressure as the containment pressure above the normal atmospheric pressure at the site, 14.4 psia.

The current licensing basis for Units 2 and 3 includes credit for containment-accident pressure (CAP) in determining available NPSH. Unit 1, due to its extended shutdown, has not previously addressed crediting CAP in determining available NPSH. The licensee discusses the history of crediting CAP for available NPSH for the BFN units in more detail in Enclosure 2 to its March 23, 2006, letter to the NRC.

To credit CAP in determining available NPSH for the design basis LOCA, licensees must demonstrate by means of conservative analyses that sufficient containment pressure will be available when required during the postulated accident when employing assumptions that overestimate suppression pool temperature and underestimate wetwell pressure.

For the short-term portion of the design basis LOCA, credit for CAP is necessary for the RHR pumps because of the high flow rates caused by the loss of pressure in the reactor vessel as a result of the recirculation discharge line break. For the long-term portion of the LOCA, the RHR pumps do not require credit for CAP. The CS pumps credit CAP in the short-term and in the long-term portions of the LOCA.

For the RHR pumps injecting into the broken loop during the short-term period of the LOCA, the containment pressure available (conservatively underestimated) is not sufficient to preclude cavitation, even when crediting reduced values of required NPSH for these pumps. These RHR pumps are predicted to cavitate for approximately 4 minutes. The NRC staff finds this acceptable for the reasons given above in the Required NPSH section.

Calculation MD-Q0999-970046 provided in Enclosure 6 to the March 23, 2006, letter, calculates the NPSH available for the RHR and CS pumps. The following table lists the pumps, their flow rates and the corresponding maximum suppression pool temperatures used in the LOCA available NPSH analyses.

Excerpts From Table 6.12 of TVA Calculation MD-Q0999-970046 Temperature and Flow Rate Combinations	
LOCA Pump/Flow Combination	Suppression Pool Temperature
CS Pumps A/B/C/D: 4125 gpm RHR A/C: 10,500 gpm each RHR B/D: 11,500 gpm each	Initial Temperature: 95 F Temperature @ 10 minutes: 155.4 degrees F
CS Pumps A/B/C/D: 4125 gpm RHR A/C: 11,500 gpm each RHR B/D: 10,500 gpm each	Initial Temperature: 95 F Temperature @ 10 minutes: 155.4 degrees F
CS Pumps A/C: 3125 gpm each, B/D: 0 RHR Pumps A/C: 6500 gpm each, B/D: 0	Maximum temperature: 187.3 degrees F
CS Pumps B/D: 3125 gpm each, A/C: 0 RHR Pumps B/D: 6500 gpm each, A/C: 0	Maximum temperature: 187.3 degrees F
CS Pumps A/C: 3125 gpm each, B/D: 0 RHR Pumps B/D: 6500 gpm each, A/C: 0	Maximum temperature: 187.3 degrees F

In order to ensure a conservative calculation for the long-term LOCA NPSH conditions, the calculated CAP is underestimated and the suppression pool temperature is overestimated. In a letter dated March 7, 2006, the licensee listed some conservative assumptions included in the calculation of wetwell conditions used in the long-term NPSH calculations. These include:

- No operator action (to initiate suppression pool cooling) for 10 minutes
- Initial reactor power 2-percent greater than RTP
- Initial reactor vessel pressure at the value
- Decay heat calculated with ANSI/ANS 5.1 with a 2σ uncertainty included
- Initial drywell temperature at the technical specification limit
- Initial suppression pool temperature at the technical specification limit
- RHR pump flow rate for suppression pool cooling: two out of four pumps (one train) available
- RHR service water inlet temperature at technical specification limit
- Bounding heat transfer coefficient for the RHR heat exchanger
- Heat sinks are credited in minimizing containment pressure. Heat sinks are not included in determining suppression pool temperature.
- Suppression pool water level at the technical specification minimum (with further drawdown of level during the accident accounted for)
- Heat transfer between suppression pool water and wetwell air space is modeled mechanistically to minimize wetwell pressure.

In addition to these assumptions is the added conservatism that all these assumptions occur simultaneously.

Initial conditions are chosen for the LOCA NPSH analyses that minimize wetwell pressure and maximize suppression pool temperature. For example, the initial drywell air temperature and the initial torus air space temperature are at their maximum values, the initial drywell and torus air pressures are at their minimum values and the initial drywell and torus relative humidities are equal to 100 percent. These values minimize the initial nitrogen which results in a lower accident pressure. The licensee determined that the worst single failure for the long-term LOCA NPSH analyses is loss of one train of emergency power.

The suppression pool level increases during the LOCA due to thermal expansion and water addition from the ECCS and FW. It decreases due to the water contained in the sprays and on the drywell floor below the elevation of the vent openings. This reduces the positive contribution of the water height above the pump suction in the available NPSH calculation (h_{static}). The NRC staff reviewed the licensee's level calculation and concluded that it correctly accounts for the geometry of the containment in predicting the amount of trapped water. The results of the licensee's calculation of the decrease in level is comparable to that done for another BWR/4 with a Mark I containment.

Required NPSH increases as the flow rate increases. The flow losses in the suction piping (h_{loss}) also increase with increasing flow rate which decreases the available NPSH. Both of these effects reduce the margin between the required and the available NPSH. The LOCA NPSH analyses assume flows of the affected CS and RHR pumps greater than or equal to the values used in the 10 CFR 50.46 LOCA analyses. This demonstrates adequate NPSH at the conditions assumed in the safety analyses.

The TS containment leakage rate (L_a) was included in the NPSH analyses since this tends to reduce the containment pressure. The containment is assumed to leak at a rate of 2-percent per day. This does not include leakage of the MS isolation valves. In a letter dated March 7, 2006, the licensee showed that MSIV leakage has only a minimal effect on the containment pressure.

The effectiveness of the RHR heat exchanger is represented by the parameter K. 1.5-percent tube plugging and a conservative fouling factor, as well as minimum RHR system and service water system flow rates, are included in the determination of K. The licensee states that:

> . . . thermal performance testing of selected heat exchangers has been performed to satisfy the requirements of GL [NRC Generic Letter] 89-13. Based on testing, appropriate inspections are performed to ensure that tube fouling does not adversely affect heat transfer performance. The heat exchanger program ensures that heat exchanger effectiveness will continue to meet the design requirements by detecting degradation before the heat transfer capabilities are adversely impacted.

The NRC staff finds the licensee's modeling of the RHR heat exchangers to be acceptable since it complies with the regulatory guidance of GL 89-13 and is conservative.

The licensee examined the effects of the drywell coolers on NPSH margin during postulated events. The licensee does not assume operation of the drywell coolers for the design basis LOCA. In a letter dated August 4, 2006, the licensee supports this assumption. The NRC staff finds the licensee's rationale for not including drywell cooler operation in the LOCA NPSH analyses acceptable.

NRC BL 96-03 recommended changes to the design of BWR ECCS suction strainers to preclude blockage due to LOCA-generated debris and other debris sources within the containment. In the July 25, 1997, letter the licensee indicated that Unit 1 was in an extended shutdown and that appropriate modifications will be implemented on Unit 1 prior to its restart. The licensee has now completed installation of new, high capacity ECCS suction strainers on Unit 1. These strainers are designed using BWROG methods approved by the NRC.

The flow resistance due to strainer blockage is based on RG 1.82. The strainer debris loading was determined in accordance with NEDO 32686P, the BWROG Utility Resolution Guidance, which was approved by the NRC staff in an SE dated August 20, 1998. The licensee states that the BFN units are essentially all reflective metallic insulation (RMI) with fibrous insulation limited to certain containment piping penetrations. The licensee conservatively assumes that the fibrous insulation available for destruction is equal to 100 percent of the fibrous insulation used in the piping penetration with the largest volume of insulation (31.7 ft^3). The amount of RMI insulation is sufficient to form a saturated debris layer around each strainer. The other debris source terms are consistent with the guidance of NEDO 32686P.

The licensee states that the only design input parameter affecting strainer head loss that changes due to the power uprate is the peak suppression pool temperature which reduces the viscosity term in the fiber head loss correlation. This is a minor effect. The pump flow rates used in the NPSH analyses are consistent with those used in the head loss calculations.

In a letter dated September 4, 1998, TVA requested credit for CAP in determining available NPSH for Units 2 and 3, as a result of the more conservative debris source term assumed with the installation of the larger passive ECCS suction strainers in response to NRC BL 96-03. The NRC approved this request by letter dated September 3, 1999. Because Unit 1 was in an extended outage, TVA did not install the large passive strainers on Unit 1 at that time.

The licensee cleans the suppression pool of each unit on a 10-year frequency. As part of the review documented in the September 3, 1999, letter, the NRC staff questioned whether the licensee's proposed frequency of cleaning the suppression pool contributed to the need for CAP for the LOCA. The NRC staff concluded at that time that the licensee's estimates of head loss across the ECCS suction strainers were reasonable and that more frequent cleaning of the suppression pool would not eliminate the need for CAP in determining available NPSH.

Because the ECCS suction strainers and the supporting analysis methods for Unit 1 are identical to those of Units 2 and 3 which the NRC staff has previously found to be acceptable, the NRC staff considers the licensee's actions responsive to the concerns raised in BL 96-03 and, therefore, BL 96-03 is closed for Unit 1.

NRC GL 97-04 requested information from licensees related to the NPSH analyses and credit for CAP in determining available NPSH. The licensee responded to GL 97-04 for all three units in a March 24, 1998, letter. This letter stated that Unit 1 was shut down and defueled without an established restart date. The letter stated that TVA would evaluate the impact of NPSH for ECCS pumps prior to restart. The NRC acknowledged completion of its review of the licensee's response for Units 2 and 3 in a June 11, 1998, letter. The licensee submitted a response to GL 97-04 for Unit 1 in a May 6, 2004, letter. The NRC staff replied to the licensee's May 6, 2004, letter in a letter dated July 27, 2006. In this letter, the NRC staff stated that the information provided in the licensee's May 6, 2004, letter will be considered as part of the review of the adequacy of the available and required NPSH of the ECCS pumps during the power uprate of Unit 1. We conclude that the information supplied by the licensee as part of the power uprate review is responsive to the GL's requests. Therefore, GL 97-04 is closed for Unit 1.

Based on the evaluation, discussed above, the NRC staff finds the licensee's LOCA NPSH analyses for Unit 1 to be acceptable for power uprate, including crediting CAP in determining available NPSH.

Postulated Accidents other than the LOCA (Special Events)

The licensee uses the term "special events" to describe the Appendix R Fire, ATWS and SBO events. These events are not DBAs.

For the SBO and ATWS events, reactor coolant inventory makeup is accomplished with the HPCI system which takes suction from the condensate storage tank rather than from the suppression pool and is therefore not affected by suppression pool conditions. The CS system is not assumed to operate during these two events. In neither of these events is there a steam discharge into the drywell. The discharge from the reactor vessel is through the MSRVs to the suppression pool.

For the Appendix R Fire, the licensee assumed a single RHR pump is injecting into the reactor vessel with flow control valves 100-percent open. The flow then returns to the suppression pool through the MSRVs. The RHR pump flow rate is governed by system resistance and the back pressure from the reactor vessel. The CS pumps are assumed not to operate during this event.

In a letter dated March 7, 2006, the licensee also addressed the effect of a stuck open relief valve on available NPSH. The licensee calculated the peak suppression pool temperature to be

154.3 degrees F. This value is lower than the suppression pool temperature that would require credit for CAP for available NPSH. Therefore, adequate available NPSH exists for the stuck open relief valve without credit for CAP.

As stated above, the pump flow for the Appendix R fire is determined from system flow resistance. The pump flows for the SBO and ATWS are assumed to be controlled by the operator. The NRC staff has examined the basis for each of these flows and finds the assumed flow rates to be acceptable, since they are flow rates the operator might reasonably choose.

The licensee considered the effects of drywell cooler operation on these events. For the Appendix R event the licensee determined that continued operation of the drywell coolers would have an adverse effect on the available NPSH. Therefore, the licensee has committed to terminate drywell cooling within two hours of entry into the safe shutdown procedure, which would be used for a shutdown due to fire. Analysis shows that this results in acceptable available NPSH for the RHR pump. For the SBO event, containment spray is assumed with restoration of ac power in accordance with the EOIs following the 4-hour coping period. The EOIs require termination of drywell coolers upon spray initiation. Therefore, the drywell coolers would not be in operation for this event. For the ATWS event, the licensee states that it is reasonable to assume the drywell coolers continue to operate. The licensee’s calculations show that margin is maintained between the available and required NPSH under these conditions.

The following table, excerpted from Table 6.12 of TVA Calculation MD-Q0999-970046, gives temperature and flow rate combinations for the special events at EPU conditions.

Excerpts From Table 6.12 of TVA Calculation MD-Q0999-970046 Temperature and Flow Rate Combinations	
Pump Flow Combination	Suppression Pool Temperature
ATWS: RHR A/B/C/D Pumps: 6500 gpm each	Maximum Temperature: 211 degrees F
Appendix R Fire: One RHR Pump (nonspecific): 9100 gpm	Maximum Temperature: 223 degrees F
Station Blackout: RHR Pumps A/C: 6500 gpm each, B/D: 0 RHR Pumps B/D: 6500 gpm each, A/C: 0	Maximum Temperature: 200 degrees F

For the SBO and ATWS events, the RHR pump flow functions to cool the suppression pool. For the Appendix R Fire event, the RHR pump injects suppression pool water into the vessel after cooling by the RHR heat exchanger. For all three special events the SHEX code was used to calculate containment conditions.

The NRC staff finds credit for CAP in determining the available NPSH for these special events is acceptable since the analyses have been done using acceptable methods and assumptions, and concludes that the pressure required for adequate NPSH margin is less than the pressure available.

Impact on Operator Response

In the March 7, 2006, letter, the licensee stated that there is no impact on the operator response, based on the existing EOIs, associated with crediting CAP at extended power conditions (which bound power uprate conditions).

The EOIs, as currently written, provide guidance to the operator to ensure containment isolation and to remain aware of the status of RHR and CS pump NPSH. The EOIs currently contain curves of suppression pool temperature as a function of pump flow with containment pressure as a parameter for both the RHR and CS pumps. There are separate curves for the RHR and the CS pumps. These curves enable the operator to ensure adequate available NPSH for these pumps.

The licensee states that Appendices to the EOIs contain operator guidance on indications of pump cavitation and possible responses. The indications of inadequate available NPSH include:

1. Suppression pool level below 10 ft,

2. System flow rate decreasing with constant valve position,

3. System flow rate or discharge pressure less than expected for the present system conditions,

4. Pump discharge pressure lower than expected and fluctuating excessively,

5. Pump motor amps lower than expected or fluctuating excessively, and

6. Pump suction pressure low (local indication).

The possible operator responses include:

1. Removing from service or throttling flow from those ECCS systems not needed to restore and maintain emergency operating instruction parameters;

2. Realigning, if possible, the suction of the CS pump(s) to the condensate storage tank. (The condensate storage tank is a nonsafety-related source of water.) The CS flow may also be reduced to maintain adequate available NPSH;

3. Using standby coolant supply (RHRSW pump injection of raw water); and

4. Considering aligning the service water system or fire protection system to the ‘A’ RHR loop.

The NRC staff considers the actions for identifying and mitigating loss of available NPSH to be acceptable since (1) they are contained in written procedures on which the operators are periodically trained, (2) there are multiple possible indications and possible mitigating actions, and (3) ECCS and suppression pool cooling functions, and hence, the proper functioning of the

ECCS and suppression pool cooling pumps, would always be a priority in terms of the operators' attention.

No operator actions are necessary to ensure sufficient containment pressure for adequate available NPSH since the pressure is a consequence of the accident. In addition, the analyses assume drywell and wetwell spray operation for the duration of the LOCA. Therefore, even if the operator does not terminate the containment spray, the analyses demonstrate that sufficient containment pressure will remain available. A Caution in the EOI directs the operator to ensure that adequate NPSH is maintained. The EOIs specify all available RHR pumps should be used for suppression pool cooling. The analyses assume a minimum number, depending on the event.

The NRC staff finds crediting CAP in determining available NPSH of the CS and RHR pumps to be acceptable with respect to the EOIs, based on adequate guidance in the emergency operating instructions, adequate indication of insufficient available NPSH, and acceptable contingency actions should loss of available NPSH occur.

Containment Integrity

Containment integrity is necessary to retain CAP. Design basis analyses, as well as the Special Events (SBO, ATWS and Appendix R Fire) assume containment integrity. This assumption is justified by the stringent requirements of the 10 CFR Part 50 and the TSs. Title 10 CFR 50.54(o) and 10 CFR Part 50 Appendix J require containment leakage rate testing of the containment structure, penetrations and isolation valves at the maximum predicted LOCA pressure. Title 10 CFR 50.55a(ii)B requires periodic inservice examination of the containment structure in accordance with the ASME Code.

Containment integrity is continuously monitored during normal operation since the containment is inerted with nitrogen gas. In a letter dated July 21, 2006, the licensee described how the makeup of nitrogen to the drywell and wetwell atmospheres serve to verify containment integrity during normal operation:

> During normal power operations, the containment is inerted with nitrogen. Per TS LCO [limiting condition for operation] 3.6.2.6, "The drywell pressure shall be maintained 1.1 psid above the pressure of the suppression chamber." Per TRM [Technical Requirements Manual] LCO 3.6.5, "When the primary containment is inerted the containment shall be continuously monitored for gross leakage by the review of the inerting system makeup requirements. Nitrogen makeup to the primary containment, averaged over 24 hours (corrected for drywell temperature, pressure and venting operations), shall not exceed 542 scfh [standard cubic ft per hour]." Per TRM Surveillance Requirement (TSR) 3.6.5.1, "When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements." The frequency of this TSR is "24 hours." Satisfying these requirements would identify any pre-existing leak in the drywell portion of containment.

The licensee also pointed out that TS 3.6.3.2 requires that the oxygen concentration in containment be maintained below 4 volume-percent during reactor power operation. Oxygen monitors in containment provide assurance that the oxygen concentration remains below this TS limit.

The licensee described measures taken to ensure that all containment penetrations are properly isolated prior to and during operation. This is repeated below.

- The primary containment air lock (TS 3.6.1.2) is a double door with limit switches on both doors that provide control room indication of door position.

- Primary containment isolation valves (TS 3.6.1.3) are controlled under plant procedures that provide strict valve controls. Aspects include valve line-up checklists, locking of specific valves, second party verification or independent verification of valve manipulations, and periodic surveillance of positions for accessible valves.

- Additionally, automatic isolation valves include position indications on the control room panels.

Conclusion

The NRC staff has reviewed the containment heat removal systems assessment provided by the licensee and concludes that the licensee has adequately addressed the effects of the proposed power uprate. The NRC staff finds that the systems will continue to meet draft GDC-10, 49, and 52 with respect to limiting the containment pressure and temperature following an LOCA and maintaining them at acceptably low levels. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to containment heat removal systems.

The NRC staff also concludes that the licensee's responses to BL 96-03 and GL 97-04 are acceptable since the licensee, by submittals made in the course of this review, has satisfied the NRC's requests in both generic communications. The NRC reviews of BL 96-03 and GL 97-04 are closed for Unit 1.

2.6.6 Secondary Containment Functional Design

Regulatory Evaluation

The secondary containment structure and supporting systems are provided to collect and process radioactive material that may leak from the primary containment following an accident. The supporting systems maintain a negative pressure within the secondary containment and process this leakage. The NRC staff's review primarily focused on the effects that the proposed power uprate may have on the pressure and temperature response and drawdown time of the secondary containment, and the impact this may have on offsite dose. The NRC's acceptance criteria for secondary containment functional design are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of an LOCA; and (2) draft GDC-10, insofar as it requires that reactor containment be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other ESFs as may be necessary, to retain functional capability for as long as the situation requires. Specific review criteria are contained in SRP Section 6.2.3.

Technical Evaluation

An increase in RTP increases the heat load on the secondary containment and affects the drawdown time of the secondary containment. The drawdown time is the time period following the start of the accident during which loss of offsite power causes loss of secondary containment vacuum (relative to atmospheric pressure), which is assumed to result in releases from the primary containment directly to the environment without filtering. The licensee addressed these issues as part of the licensee's conversion to the AST. The AST, including the secondary containment drawdown time, was analyzed at EPU conditions which bound conditions at power uprate. The NRC staff found the licensee's proposed conversion to the AST to be acceptable in a September 27, 2004, letter to the licensee.

The licensee indicated that increased power does not change the volume or the alignment of the secondary containment, the normal operating conditions of the secondary containment atmosphere, nor the alignment, actuation, or the operation of the Standby Gas Treatment System (SGTS). Therefore, the ability to achieve a negative draw down pressure in the secondary containment is not affected by the power uprate. The secondary containment drawdown time is verified by performance of periodic surveillance. The testing is performed to verify secondary containment drawdown and integrity in compliance with the requirements of the TSs to ensure the design basis requirements are met.

Conclusion

The NRC staff has reviewed the licensee's assessment related to the secondary containment pressure and temperature transient and the ability of the secondary containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment. The NRC staff concludes that the secondary containment and associated systems will continue to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment following implementation of the proposed power uprate. Based on this, the NRC staff also concludes that the secondary containment and associated systems will continue to meet the provisions of draft GDC-10, 40, and 42. Therefore, the NRC staff finds the secondary containment functional design acceptable for power uprate.

2.6.7 Additional Review Areas (Containment Review Considerations)

Hardened Wetwell Vent

Regulatory Evaluation

As a mitigation measure, a reliable wetwell vent provides assurance of pressure relief through a path with significant scrubbing of fission products and can result in lower releases even for containment failure modes not associated with pressurization (i.e., liner meltthrough). In the 1980's as a part of a comprehensive plan for closing severe accident issues, the NRC staff undertook a program to determine if any actions should be taken, on a generic basis, to reduce the vulnerability of BWR Mark I containments to severe accident challenges. At the conclusion of the Mark I Containment Performance Improvement Program, the NRC staff identified a number of plant modifications that substantially enhance the plants' capability to both prevent and mitigate the consequences of severe accidents. The improvements that were recommended included an improved hardened wetwell vent capability. Consistent with the recommendations of GL 89-16, *Installation of a Hardened Wetwell Vent*, the licensee installed such a vent on Browns Ferry.

Technical Evaluation

Unit 1 had already been shutdown when GL 89-16 was issued. As indicated in the UFSAR, the consequences of several beyond DBA scenarios are more severe than the accidents considered in UFSAR. The primary containment pressure during these accidents is estimated to exceed its design capacity. Thus, the primary containment fails, potentially to the environment as well. The hardened wetwell vent (HWWV) provides an emergency primary containment vent path to prevent, or at least slow down, the buildup of potentially damaging pressure within the primary containment.

The hardened vent design criterion is to maintain containment design pressure with the reactor at 1-percent of RTP. The licensee stated that the current design of the HWWV was based on 1.05-percent of 3293 MWt OLTP. The design criterion is 1 percent and is designed to be operational during an SBO. Therefore, the HWWV will satisfy its design basis at the proposed power uprate conditions.

Conclusion

The NRC staff has reviewed the licensee’s assessment related to the ability of the containment to maintain the design pressure with the reactor at 1-percent RTP. The NRC staff concludes that the hardened wetwell vent can maintain at power uprate conditions.

Containment Isolation

Regulatory Evaluation

The NRC’s acceptance criteria for containment isolation are based on draft GDC-52 insofar as the containment isolation function must be protected by redundant valving and associated apparatus.

Technical Evaluation

The licensee stated that the system designs for containment isolation are not affected by power uprate and the capabilities of isolation actuation devices to perform under normal and post-accident conditions are acceptable. The licensee reviewed the AOV and SOV parameters (temperature, pressure and flow) and no changes to the functional requirements of any AOV and SOV were identified as a result of operating at the EPU conditions. The licensee reviewed the MOV process parameters (temperature, pressure and flow) and no significant changes to the functional provisions of the GL 89-10 program for MOVs were identified as a result of operating at EPU conditions. The licensee's pump and valve program is addressed elsewhere in this SE. The NRC staff did not identify any concerns regarding containment isolation for power uprate conditions.

An important aspect of the effect of containment accident conditions on containment isolation is addressed by GL 96-06, *Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions*. One of the topics addressed in GL 96-06 is thermally-induced overpressurization of isolated water filled piping sections in containment which could jeopardize the ability of accident mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage. The licensee indicated that its evaluation was accomplished using the peak drywell temperature for a steam line break inside containment. The licensee's analysis concluded that the equipment and containment remain within their design allowable values.

The NRC staff reviewed the licensee's assessment related to this aspect of GL 96-06 and concludes that the licensee has adequately addressed the issue of thermally-induced overpressurization of the affected piping in containment under uprated conditions.

Conclusion

Based on the above, the system designs for containment isolation capabilities are not adversely affected by the power uprate and continue to meet the provisions of draft GDC-52. Therefore, the NRC staff finds that containment isolation remains acceptable for power uprate conditions.

2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

The NRC staff reviewed the control room habitability system and control building layout and structures to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases. A further objective of the NRC staffs review was to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. The NRC staff's review focused on the effects of the proposed power uprate on radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination. The NRC's acceptance criteria for the control room habitability system are based on GDC-19, insofar as they require that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. Specific review criteria are contained in SRP Section 6.4 and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

For control room habitability, the NRC staff reviewed the control room ventilation system and control building layout and structures, as described in the applicant's safety analysis report (SAR), the applicant's response to GL 2003-01, dated December 8, 2003, which defines the control room habitability zone, and the analysis presented to the NRC in Enclosure 4 to the letter dated June 28, 2004, regarding the control room habitability aspect of the EPU and the licensee's AST license amendment. As stated in the June 28, 2004, submittal, the licensee performed AST analyses for the four DBAs that could potentially result in significant control room and offsite doses. The four accidents included the LOCA, the MSLB accident, the refueling accident, and the CRD accident. The NRC staff's independent calculations and review agree that the licensee's analyses demonstrate that, using AST methodologies, post-accident control room and offsite doses remain within regulatory acceptance limits. The

NRC staff also noted that sources, locations and quantities of toxic gases are not changed by the power uprate, thus there is no impact of the uprate on mitigation of toxic gas.

The NRC staff is currently working toward resolution of generic issues related to control room habitability, in particular, the validity of control room inleakage rates assumed by licensees in analyses of control room habitability. The NRC staff issued GL 2003-01, Control Room Habitability. TVA responded to this GL, by letter dated December 8, 2003. In this response, TVA reported that inleakage testing using the ASTM tracer gas methodology yielded a control room unfiltered inleakage rate of only 600 cfm. This value is approximately 84-percent less than the 3717 cfm assumed in the BFN design and licensing basis, a conservative situation. Although the TVA response to the GL is still under review, the NRC staff previously determined, in its review of the AST implementation, that there is reasonable assurance that the BFN control room will be habitable during DBAs. The NRC staff previously found that the AST implementation amendment could be approved before the final resolution of the generic issue. Because the DBA radiological consequences are bounding for the 5-percent power uprate, the same reasoning may be applied to find the approval of this amendment request acceptable before the final resolution of the generic issue. The NRC staff's acceptance of TVA's unfiltered inleakage assumption for the purposes of the AST implementation and the proposed 5-percent power uprate does not establish that the NRC staff has found the December 8, 2003, response adequate. The NRC staff will respond to TVA's GL response under separate correspondence.

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the proposed power uprate on the ability of the control room habitability system to protect plant operators against the effects of accidental releases of toxic and radioactive gases. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from the proposed uprate. The adequacy of the licensee's resolution of generic issues affecting control room habitability is being reviewed as part of GL 2003-01. Based on this, the NRC staff concludes that this review combined with the satisfactory completion of GL 2003-01, the control room habitability system will continue to meet the requirements of GDC-19. Therefore, the NRC staff finds the control room habitability acceptable for power uprate conditions.

2.7.2 Engineered Safety Feature Atmosphere Cleanup

ESF atmosphere cleanup systems are designed for fission product removal in post-accident environments. These systems generally include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., standby gas treatment systems and emergency or post-accident air-cleaning systems) for the fuel-handling building, control room, shield building, and areas containing ESF components. For each ESF atmosphere cleanup system, the NRC staff's review focused on the effects of the proposed power uprate on system functional design, environmental design, and provisions to preclude temperatures in the adsorber section from exceeding design limits.

Regulatory Evaluation

The NRC's acceptance criteria for ESF atmosphere cleanup systems are based on (1) GDC-19 and 10 CFR 50.67, insofar as they require that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident; (2) draft GDC-67, 68, and 69, insofar as they require that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and (3) draft GDC-17, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs, and postulated accidents. Specific review criteria are contained in SRP Section 6.5.1.

Technical Evaluation

The function of the ESF atmosphere cleanup system is to mitigate the consequences of postulated accidents by removing from the atmosphere radioactive material that may be released in the event of an accident. ESF atmosphere cleanup systems should be designed so that they can operate after a DBA and can retrain radioactive material after a DBA. The system has provisions to prefilter air, remove moisture, and meet appropriate surveillance test requirements for filter system performance.

The ESF atmospheric cleanup system at Unit 1 is the SGTS. The SGTS is designed to maintain secondary containment at a negative pressure and to filter the exhaust air for removal of fission products potentially present during abnormal conditions. By limiting the release of airborne particulate and halogens, the SGTS limits off-site dose following a postulated DBA. As discussed in the analysis presented in the PUSAR regarding the SGTS and the licensee’s AST license amendment, the acceptability of the SGTS was determined by reviewing the dose consequences of DBAs.

The NRC staff determined that the capability of the SGTS is unaffected by power uprate because the specified primary and secondary leak rates are not affected. Also, the High Efficiency Particulate Air filters have sufficient design margin to accommodate additional fission product loading without restricting flow rate. The carbon adsorber removal efficiency for radioiodine is also not affected by power uprate and the carbon filter banks have sufficient capacity to adsorb the additional source term.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed power uprate on the ESF atmosphere cleanup systems. The NRC staff concludes that the licensee has adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed power uprate, and the NRC staff further concludes that the ESF atmosphere cleanup systems will continue to provide adequate fission product removal in post-accident environments following implementation of the proposed power uprate. Based on this, the NRC staff concludes that the ESF atmosphere cleanup systems will continue to meet the requirements of GDC-19, draft GDC-17, 67, 68, and 69; and 10 CFR 50.67. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the ESF atmosphere cleanup systems.

2.7.3 Control Room Area Ventilation System

Technical Evaluation

The NRC staff reviewed the licensee's evaluation which determined the effect of power uprate on process temperature and electrical heat load changes on the Control Room Heating, Ventilation, and Air Conditioning (HVAC) System. The evaluation considered present HVAC equipment capacity, area heat and electrical load changes, and area temperature changes.

The Control Bay HVAC systems serve the three floors in the control bay and the six shutdown electrical board rooms in the Reactor Building immediately adjacent to, and normally entered from, the control bay. There are several separate subsystems serving these areas. Included are the Control Bay, Units 1 and 2 Control Room , Units 1 and 2 computer rooms, electrical board rooms, auxiliary instrument rooms, switchyard relay room, and the Unit 3 Control Room. The Unit 3 work with this area is heated and cooled with a separate air supply system, but it is not thermostatically controlled. The air supply systems for three areas serve a group of rooms with only cooling. These areas are the Unit 1 Electric Board Rooms, Unit 2 Electric Board Rooms, and the Unit 3 Electric Board Rooms.

Each cable spreading room is ventilated by one 100-percent capacity fresh-air supply fan. Two 100-percent exhaust fans serve both of these rooms. These two rooms serve all three BFN units. The Control Bay does not contain steam cycle process equipment, but rather it primarily contains the electrical and instrumentation equipment necessary to control the process equipment.

The NRC staff finds that there is no heat dissipation increase by this electrical and instrumentation equipment due to power uprate operation. The power uprate does not impact the design conditions of the system evaluated and the present HVAC equipment capacity remains adequate.

2.7.4 Spent Fuel Pool Area Ventilation System

Regulatory Evaluation

The function of the SFP area ventilation system (SFPAVS) is to maintain ventilation in the SFP equipment areas, permit personnel access, and control airborne radioactivity in the area during normal operation, AOOs, and following postulated fuel-handling accidents (FHAs). The NRC staff's review focused on the effects of the proposed power uprate on the functional performance of the safety-related portions of the system. The NRC's acceptance criteria for the SFPAVS are based on draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents to the environment. Specific review criteria are contained in SRP Section 9.4.2.

Technical Evaluation

As indicated in Enclosure 12 of the June 28, 2004, submittal, BFN does not have a separate SFPAVS. In a letter dated March 7, 2006, the licensee stated that the SFP area is ventilated by the reactor building ventilation system. The general Reactor Building areas are heated, cooled, and ventilated during normal and shutdown operation by a once-through air system. The ventilation system provides 100-percent makeup air. The reactor building ventilation air is supplied to the reactor building spaces via supply fans, drawn through the building by roof mounted exhaust fans, and then directly exhausted to the atmosphere via ductwork monitored for radiation.

The primary power uprate impact on SFP is an increase in the pool decay heat loads following discharge of spent fuel in refueling outages. The licensee indicated that operation at higher power levels requires a higher burn-up of fuel, and therefore a greater heat load will accompany the discharge of this fuel in an outage. There are no process temperature changes or electrical load changes associated with the SFPCS that result from power uprate operation, and the existing SFP design temperature limits remain unchanged for power uprate conditions. However, the temperature of the SFP will increase and makeup water demand will increase, but both parameters remain within existing acceptance criteria.

The licensee evaluated a loss of SFP cooling for both the batch and full core offload scenarios. Maximum boil off rates remain well within pool inventory make-up capacity and design flows, temperatures, and pressures are adequate for rejecting the increased SFP heat load. The TS limit of 150 degrees F is not being changed, therefore the SFP will be maintained below this temperature during power uprate operation in the same manner it is currently.

As individual SFP system components and design limits are not affected by power uprate, no SFP system modifications are required to support power uprate. Any increases in SFP temperature remain within the capacity of the ventilation system, therefore, the NRC staff finds that the proposed power uprate does not impact the design conditions of the system, and the present HVAC equipment capacity remains adequate.

Conclusion

The NRC staff has reviewed the licensee’s assessment of the effects of the proposed power uprate on the SFPAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system’s capability to maintain ventilation in the SFP equipment areas, permit personnel access, control airborne radioactivity in the area, control release of gaseous radioactive effluents to the environment, and provide appropriate containment. Based on this, the NRC staff concludes that the SFPAVS will continue to meet the requirements of draft GDC-70. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the SFPAVS.

2.7.5 Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems

Technical Evaluation

The NRC staff reviewed the licensee’s evaluation which determined the effect of power uprate process temperature and electrical heat load changes on the Radwaste Building Ventilation System. The review considered present HVAC equipment capacity, area heat and electrical load changes, and area temperature changes.

As uprate does not impact the design conditions of the system evaluated and the present HVAC equipment capacity remains adequate. The power uprate results in no process temperature changes in the Radwaste Building. There is an increase in the volume of liquid radwaste which must be processed as a result of power uprate operations, but the individual batch quantities of water being processed will only increase in number, not in process temperature. There will also be no additional work required for the processing of these additional batches, therefore there are also no electrical load changes.

The NRC staff finds that with no additional heat load from either the liquid radwaste volume or the work required to process it, there is no additional load on the radwaste building HVAC equipment resulting from power uprate operations.

2.7.6 Engineered Safety Feature Ventilation System

Regulatory Evaluation

The function of the ESF ventilation system (ESFVS) is to provide a suitable and controlled environment for ESF components following certain anticipated transients and DBAs. The NRC staff's review for the ESFVS focused on the effects of the proposed power uprate on the functional performance of the safety-related portions of the system. The NRC staff's review also covered (1) the ability of the ESF equipment in the areas being serviced by the ventilation system to function under degraded ESFVS performance; (2) the capability of the ESFVS to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components (e.g., storage batteries and stored fuel); and (3) the capability of the ESFVS to control airborne particulate material (dust) accumulation. The NRC's acceptance criteria for the ESFVS are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of an LOCA; (2) draft GDC-24 and 39, insofar as they require onsite and offsite electric power systems be provided to permit functioning of the ESFs and protection systems; and (3) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.5.

Technical Evaluation

The power uprate impacts on the air-conditioning and ventilation systems in the control bay and the electrical board rooms that are functionally part of the control bay spaces are discussed in Section 2.7.3. The uprate impacts on the 4-kV shutdown boards, the diesel generator building (DGB), and the RHR pump and CS pump areas in the reactor building are addressed below.

The NRC staff reviewed the licensee’s evaluation which determined the effect of power uprate process temperature and electrical heat load changes on the DGBs, the 4-kV shutdown board rooms, and RHR/CS pump spaces. The review considered present HVAC equipment capacity, area heat and electrical load changes, and area temperature changes.

The Unit 3 electric board rooms are cooled by redundant air-conditioning units. The DGB HVAC systems are designed to maintain the required environmental conditions for safety-related equipment located in the Units 1 and 2 and the Unit 3 DGB. Ventilation cooling and fume removal from each of the eight (8) (DG) rooms is provided by one of two redundant exhaust fans (A and B per DG) with associated room inlet and outlet and fan discharge motor operated dampers. These fans discharge into a common exhaust plenum open to the atmosphere for each respective building. The RHR and CS pumps are located in the basement rooms of the Reactor Building. The heat loss from the motors, pumps, and piping is removed by air-cooling units. The air-cooling units are designed to maintain the air at 148 degrees F when the unit is supplied with 95 degrees F cooling water. An equipment area air-cooling unit starts automatically when an RHR pump (or a CS pump) in that compartment starts. The air-cooling units also start automatically when compartment temperatures approach 100 degrees F.

Due to power uprate, there are minimal area heat load impacts in the CS and RHR rooms. The temperature increase in the CS pump rooms is less than 3 degrees F. The temperature increase in the RHR pump rooms is less than 2 degrees F. Increase in suppression pool temperature increases the suppression piping heat loads. The torus space temperature also increases due to the increase in suppression pool temperature. The torus space adjoins the RHR and CS rooms and will increase the wall heat transfer load into these rooms. The heat rejection capacity for these room coolers was reviewed, and the coolers are deemed adequate for uprated conditions in the RHR and CS rooms.

As a result of the review of the primary heat loads and power uprate impact, the NRC staff finds that the present ventilation systems are adequate to support operation at power uprate conditions.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed power uprate on the ESFVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed uprate on the ability of the ESFVS to provide a suitable and controlled environment for ESF components. The NRC staff further concludes that the ESFVS will continue to assure a suitable environment for the ESF components following implementation of the proposed uprate. The NRC staff also concludes that the ESFVS will continue to suitably control the release of gaseous radioactive effluents to the environment following implementation of the proposed uprate. Based on this, the NRC staff concludes that the ESFVS will continue to meet the provisions of draft GDC-24, 39, 40, 42, and 70. Therefore, the NRC staff finds the proposed uprate acceptable with respect to the ESFVS.

2.8 Reactor Systems

The nuclear, thermal-hydraulic evaluations, transients analyses (core-related) are based on the NRC staff review of the Supplemental Reload LTR for Unit 1 Reload 6 Cycle 7 dated May 2006. This Supplemental Reload Licensing Report (SRLR) includes results of the Unit 1 cycle-specific core analyses, and the transient and accident analyses performed for the core design and for the core operating during Cycle 7.

TVA provided a new SRLR for 105-percent specifically addressing the safety limit minimum critical power ratio (SLMCPR) on January 31, 2007. The NRC staff has reviewed this report and confirmed that the analyses remain applicable for operation throughout the upcoming operating cycle at the 105-percent power level.

In general, the licensee's plant-specific engineering evaluations supporting the power uprate were performed in accordance with guidance contained in the GE ELTR1. This LTR was previously reviewed and approved by the NRC staff. For some items, bounding analyses and evaluations provided in GE ELTR2 were cited. This LTR was also reviewed and approved by the NRC staff. The ELTR2 generic evaluations assume (a) a 20-percent increase in the thermal power; (b) an increase in operating dome pressure up to 1095 psia; (c) a reactor coolant temperature increase to 556 degrees F; and (d) a steam and FW flow increase of about 24 percent.

2.8.1 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs; (2) fuel system damage is never so severe as to prevent control rod insertion when it is required; (3) the number of fuel rod failures is not underestimated for postulated accidents; and (4) coolability is always maintained. The NRC staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC’s acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits; and (3) draft GDC-37, 41, and 44, insofar as they require that a system to provide abundant emergency core cooling be provided to prevent fuel damage following an LOCA. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Fuel system design at Unit 1 is described in Section 3.2 of the UFSAR. Section 5.1 of ELTR2 states that no change is required to the basic fuel design to achieve the uprated power level or to maintain the safety margin. GE fuel design, up through GE-14 fuel, was approved for previous EPU operation in other BWRs. The Unit 1 fuel for Cycle 7 consists of GE-13 and GE-14 design. The core thermal-hydraulic design and fuel performance characteristics are evaluated for each reload fuel cycle.

Fuel Design and Operation

The power distribution in the core is changed to achieve increased core power, while limiting the Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR), and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) in any individual fuel bundle to be within its operating limits as defined in the core operating limits report (COLR).

The OLTP core for Unit 1 consists only of GE fuel types. The Cycle 7 core primarily consists of fresh fuel and uses 564 fresh GE-14 and 108 fresh GE-13 fuel assemblies. The design also includes 36 previously irradiated GE-14 and 56 previously irradiated GE-13 fuel assemblies, which were discharged from the Unit 2 Cycle 13 core, as described in the SRLR for Unit 1 Cycle 7.

Thermal Limits Assessment

NRC's acceptance criteria require that the reactor core and the associated control and instrumentation systems be designed with appropriate margin to ensure that the specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory or safety limits are not exceeded for a range of postulated events (transients and accidents). The operating and safety limits (i.e., operating limit MCPR, safety limit MCPR, MAPLHGR and the LHGR operating limits) are cycle dependent and as such will be established or confirmed at each reload.

The SLMCPR ensures that 99.9-percent of the fuel rods are protected from boiling transition during steady-state operation. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as result of an AOO. NRC staff experience with several power uprates has shown that the change in OLMCPR resulting from an EPU is small. The OLMCPR will be determined for plant cycle-specific core design parameters using approved methods, as described in Sections 5.3.2 and 5.7.2.1 of ELTR1 and Section 3.4 of ELTR2. Because the licensee has used approved methods in the SRLR for Cycle 7, and will continue to use approved methods to evaluate these parameters, this is acceptable to the NRC staff. As required by the ELTR1 and ELTR2, the licensee will perform plant cycle-specific reload analysis to demonstrate that the SLMCPR and OLMCPR are appropriate for establishing the EPU thermal limits.

The licensee stated in its September 22, 2006, submittal that all safety analyses for operation at 105-percent OLTP were acceptably bound by previous analyses performed assuming Cycle 7 operation at 120-percent OLTP. While the NRC staff concluded, in most cases, that this assumption was acceptable, several concerns were identified with regard to prolonged changes in operating strategies that could affect the SLMCPR in a nonconservative manner. Specifically, the NRC staff was concerned that changes in control rod pattern and core flow could change the core power distribution and effect an increase in the SLMCPR. Increases in SLMCPR would be possible through extended changes in operating strategy in two ways:

(1) insertion of additional control rods could cause the axial power shape of individual rods and bundles to become more outlet-peaked, and

(2) reduction of power at the center of the core could result in a flatter radial power distribution, placing more control rods near boiling transition in the limiting SLMCPR scenario.

Therefore, the NRC staff requested confirmation that the SLMCPR values provided in 2006 for operation at 120-percent OLTP for the entire cycle, and approved by the NRC staff in a SE dated February 6, 2007, would remain bounding for operation at all licensed power levels. By letter dated November 6, 2006, the licensee provided a recalculation of the SLMCPR for the Unit 1 Cycle 7 operation at 105-percent OLTP that demonstrated that the SLMCPR calculation remained bounding. The SLMCPR re-analysis performed by GE, the licensee's fuel vendor, specifically considered axial power shapes, and concluded the following:

> For the limiting bundles, the fuel axial power shapes in the SLMCPR analysis were examined to determine the presence of [potentially limiting] axial power shapes. . . . These power shapes were not found; therefore, no power shape penalties were applied to the calculated Browns Ferry Unit 1 Cycle 7 SLMCPR values.

The NRC staff finds this acceptable because its supporting analysis was performed in accordance with the NRC-approved methodologies. Therefore, no additional conservatism is necessary beyond the currently licensed SLMCPR values with regard to axial power shaping resulting from potential changes in operating strategy.

The fuel vendor determined that the minimum core flow SLMCPR calculation performed at 81-percent core flow and rated power condition was limiting as compared to the rated core flow and rated core power condition. GE determined that the minimum core flow SLMCPR calculation is bounding on the basis that it was performed using a limiting control rod pattern as compared to a nominal control rod pattern. The NRC staff finds that the use of a limiting control rod pattern at a limiting state point is a duly conservative approach that provides an acceptable SLMCPR value. The NRC staff also confirmed that the SLMCPR analyzed by GE for this purpose was less than the SLMCPR approved by the NRC staff on February 6, 2007, ensuring that the approved SLMCPR is more conservative. Therefore, the NRC staff finds that no additional conservatism for the SLMCPR values is necessary for changes in power distribution due to operation at 105-percent OLTP.

In consideration of the fact that the SLMCPR values require no additional conservatism to ensure that, during normal operation, 99.9 percent of the fuel rods would be expected not to undergo boiling transition throughout the upcoming operating cycle, the NRC staff finds that the SLMCPR values approved for Unit 1 Cycle 7, remain acceptable in light of potential changes in operating strategy to support operation at 105-percent OLTP.

The MAPLHGR operating limit is based on the most limiting LOCA conditions, and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, GE performs LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload licensees confirm that the MAPLHGR operating limit for each reload fuel bundle design remains applicable. As discussed in Section 2.8.5.6, the licensee performed an LOCA evaluation for the Unit 1 Cycle 7 core designed for EPU operation, as submitted in the SRLR. The licensee stated that the LOCA analysis showed no change in the MAPLHGR or the LHGR limits for normal operation.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed uprate on the fuel system design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed uprate on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and AOOs, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, draft GDC-6, 37, 41, and 44 following implementation of the proposed uprate. Therefore, the NRC staff finds the proposed uprate acceptable with respect to the fuel system design.

2.8.2 Nuclear Design

Regulatory Evaluation

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NRC staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits during any condition of normal operation, including the effects of AOOs; (2) draft GDC-8, insofar as it requires that the reactor core be designed so that the overall power coefficient in the power operating range shall not be positive; (3) draft GDC-7, insofar as it requires that the reactor core be designed to ensure that power oscillations, which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed; (4) draft GDC-12, insofar as it requires that instrumentation and controls are provided as required to monitor and maintain variables within prescribed operating ranges; (5) draft GDC-14 and 15, insofar as they require that the protection system be designed to initiate the reactivity control systems automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and to initiate operation of ESFs under accident situations; (6) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits; (7) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits; (8) draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and (9) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the RCPB or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The nuclear design for Unit 1 is described in Section 3.6 of the UFSAR. The higher core energy requirements of a power uprate may affect the hot excess core reactivity and can also affect operating shutdown margins. Based on experience with previous plant-specific power uprate submittals, the required hot excess reactivity and shutdown margin can typically be achieved for power uprates through the standard approved fuel and core reload design process. Plant shutdown and reactivity margins must meet NRC-approved limits established in GESTAR-II on a cycle-specific basis and are evaluated for each plant reload core. Additional hot excess reactivity and shutdown margin analyses are not specifically required for the uprate.

In a letter dated March 7, 2006, the licensee validated whether the ELTR1 assumptions regarding nuclear design remained valid such that this area can be generically dispositioned. The licensee's review confirmed that the Unit 1 reactivity characteristics are consistent with the generic description discussed in the ELTR1, and that the shutdown margin for each uprated reload core will be evaluated prior to power uprate implementation.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed uprate on the nuclear design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed uprate on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable provisions of draft GDC-6, 7, 8, 12, 14, 15, 27, 28, 29, 31, and 32. Therefore, the NRC staff finds the nuclear design is acceptable for power uprate conditions.

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The review also covered core thermal-hydraulic stability under normal operation and ATWS events. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits during any condition of normal operation, including the effects of AOOs, and (2) draft GDC-7, insofar as it requires that the reactor core, together with reliable controls, ensure that power oscillations, which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed. Specific review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Thermal and hydraulic design of BF-1 is described in Section 3.7 of the UFSAR. Consistent with Section 3.2 of ELTR1 and ELTR2, the evaluation for thermal hydraulic stability was performed for Cycle 7, as submitted in the SRLR.

As discussed in the SE dated December 26, 2006, Unit 1 will implement BWROG Long Term Stability Solution Option III using the Oscillation Power Range Monitor (OPRM) as described in NEDO-31960-A and NEDO-31960, Supplement 1, "BWROG Long-Term Stability Solution Licensing Methodology." TVA stated that it would implement the Option III methodology as an integrated part of an advanced digital power range neutron monitoring (PRNM) upgrade using

GE's Nuclear Measurement, Analysis, and Control (NUMAC) equipment. Stability Long Term Solution Option III consists of hardware and software that provide for reliable, automatic detection and suppression of stability related power oscillations. Hardware to implement the OPRM Upscale trip, for the proposed new TS, is housed in the same chassis as the average power range monitor (APRM) hardware, and the OPRM Upscale trip is considered a sub-function of the APRM system.

The PRNM system, which includes four APRM subsystem and the OPRM function, is described in detail in GE Topical Report NEDC-32410P-A, Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option Stability Functions. The OPRM trips that will be enabled for Unit 1 Cycle 7 are the licensing basis Period Based Detection Algorithm, as well as for the Growth Rate Algorithm, and Amplitude Based Algorithm defense-in-depth features. The algorithms for the LTSS Option III solution are described in NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications. The hardware/software for Unit 1 will also contain the Confirmation Density (CD) algorithm, however, the trip for this feature will not be enabled. Global Nuclear Fuel, a subsidiary of GE, is the only fuel vendor responsible for the Unit 1 core design and licensing analysis and no other fuel vendor is involved.

The OPRM amplitude setpoint calculation includes three components as defined in NEDO-32465-A. The calculation for hot channel oscillation magnitude is performed using the approved GE methodology, and the generic DIVOM calculations performed in NEDO-32465-A used the earlier TRACG02 version and pre-PANAC11 neutronic method. GE has performed an evaluation comparing the use of TRACG04-PANAC11 versus TRACG01-PANAC10 in the calculation of DIVOM slopes and determined that results are essentially the same. Cycle-specific setpoint calculations are now performed to determine the operating MCPR needed to protect the SLMCPR for the various OPRM amplitude setpoints. The Option III trip is armed only when plant operation is within the Option III trip-enabled region. The Option III trip-enabled region is defined as the region on the power/flow map with power < 30-percent OLTP and core flow ≤ 60-percent rated core flow.

Unit 1 uses the BWROG Interim Corrective Action (ICA) stability regions as the backup stability protection method when the OPRM system is declared to be inoperable. These regions are confirmed on a cycle-specific basis by performing backup stability protection calculations in accordance with the guidance provided in OG02-0119-260, Backup Stability Protection (BSP) for inoperable Option III Solution, dated July 17, 2002. The GE ODYSY code is used for the calculation of decay ratios based on statepoint and neutronic data from PANAC11 and TGBLA06.

An on-site audit of the LTSS implementation and ATWS instability of Units 1, 2 and 3 was performed by the NRC staff and its consultant from Oak Ridge National Laboratory on August 8, 2006. The NRC staff found that:

1. TVA plans to implement Option III using Detect and Suppress Solution/Confirmation Density (DSS/CD) NUMAC hardware, but CD portion of algorithm will not be armed. DSS/CD hardware implements all three Option III scrams as defense in depth. By disabling the CD scram, the hardware implementation in Unit 1 will revert to the standard Option III, where the licensing scram is the Period Based Detection Algorithm (PBDA) signal. This is an acceptable implementation.

2. Unit 1 will have the option to arm the CD scram function without having to upgrade hardware because the DSS/CD EPROMS are already installed and tested. DSS/CD is a new approved long term stability solution, similar in many respects to Option III. It uses the Option III PBDA with more restrictive parameter setting. To prevent spurious scrams, DSS/CD requires confirmation by a large number of OPRM cells.

3. The NRC staff reviewed a transient scenario in the Browns Ferry control room simulator that involved unstable power oscillations. The training staff was familiar with stability problems and actions to prevent them. As with the real plant, the simulator has Option III installed as the licensed solution. Note that this is a fairly slow-developing instability, so Option III detected the number of confirmation, but had to wait until the amplitude grew large enough to reach the scram setpoint.

4. The TSs rely on the BWROG ICAs when the OPRM system is unavailable. It is acceptable that BFN uses generic (step-wise) BWROG regions for their ICAs, which are verified for adequacy every reload. Under reduced FW temperature conditions, the cycle-specific BFN exclusion regions are slightly larger than the step-wise generic ICAs and the plant computer displays a combination of the two (step-wise and cycle-specific) regions to define a conservative exclusion region.

5. The OPRM trip setpoints and regions of ICAs will be specified either in the licensee operating instruction or in the core operating limits report.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed power uprate on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods; (2) is consistent with proven designs; (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs; and (4) is not susceptible to thermal-hydraulic instability. The NRC staff further concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the hydraulic loads on the core and RCS components. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the provisions of draft GDC-6 and 7 following implementation of the proposed power uprate.

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covered the functional performance of the CRDS to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of an LOCA; (2) draft GDC-26, insofar as it requires that the protection system be designed to fail into a safe state; (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits; (4) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits; (5) draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; (6) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the RCPB or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (7) 10 CFR 50.62(c)(3), insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system, and that the ARI system have redundant scram air header exhaust valves. Specific review criteria are contained in SRP Section 4.6.

Technical Evaluation

The CRD System is described in Section 3.4 of the UFSAR. The CRD system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The scram, rod insertion and withdrawal functions of the CRD system depend on the operating reactor pressure and the pressure difference between the CRD system hydraulic control unit and the reactor vessel bottom head pressure. Unit 1 has installed an ARI system which is diverse from the RPS and the Unit 1 ARI system has redundant scram air header exhaust valves to improve reliability.

The Unit 1 higher reactor dome pressure has little effect on scram time and the scram time performance relative to current plant operation is not significantly affected. Therefore, the current TS scram requirements are still valid. In Section 2.5 of the PUSAR, the licensee indicated that for CRD insertion and withdrawal, there will be a minimum pressure of 250 psid between the hydraulic control unit and the vessel bottom head. The licensee confirmed that sufficient capability exists to support this differential. Additionally, the automatic operation of the system flow control valve maintains the required drive water pressure. This ensures that system remains capable of compensating for expected pressure increases.

The CRD system capability to sustain any single malfunction without causing a reactivity transient is unaffected by the power uprate. Two independent reactivity control systems, CRD system and SLCS, are still provided. The capability of either system to make the core subcritical under any conditions remains available at uprated conditions. Control rod worth limits, which include considerable margin, are unaffected.

As described in Section 2.5.3 of the PUSAR, the licensee reviewed increases in vessel pressure and temperature as a result of power uprate. The analyses confirmed that, for these conditions, the CRD system integrity is assured given normal and abnormal pressure events as well as all applicable stress intensity limits which are governed by fatigue.

The CRD system was generically evaluated in ELTR1 and ELTR2. The generic evaluation concluded that the CRD systems for BWR/2-6 types are acceptable for power uprates as high as 20-percent above the original rated power. The NRC staff concluded that no additional plant-specific calculations are required beyond confirmatory evaluation. In Section 2.5 of the PUSAR, the licensee confirmed that the generic evaluation for the scram time response, CRD positioning, CRD cooling and CRD integrity is applicable to Unit 1. The licensee also determined that no modifications or changes are required as a result of power uprate. As the licensee's analyses are acceptably bounded by the generic evaluations in the uprate TRs, the NRC staff finds that the CRD system remains consistent with draft GDC 27 and 28. This is due to the CRD system remaining one of two systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed uprate on the functional design of the CRDS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed uprate on the system and demonstrated that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed uprate. The NRC staff further concludes that the licensee has demonstrated that sufficient cooling exists to ensure the system's design bases will continue to be followed upon implementation of the proposed uprate. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the provisions of draft GDC-26, 27, 28, 29, 31, 32, 40, and 42, and 10 CFR 50.62(c)(3) following implementation of the proposed uprate. Therefore, the NRC staff finds the proposed uprate acceptable with respect to the functional design of the CRDS.

2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the RPS. The NRC staff's review covered relief and safety valves on the MSLs and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (2) draft GDC-33, 34, and 35, insofar as they require that the RCPB be designed to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating type failures is minimized. Specific review criteria are contained in SRP Section 5.2.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Nuclear system pressure relief system is discussed in Section 4.4 of UFSAR. The safety/relief valves (SRVs) provide overpressure protection for the NSSS, preventing failure of the nuclear system pressure boundary and uncontrolled release of fission products. Unit 1 has 13 SRVs. The SRV set points are established to provide the overpressure protection function while ensuring that there is adequate pressure difference (simmer margin) between the reactor operating pressure and the SRV actuation set points. The SRV set points are also selected to be high enough to prevent unnecessary SRV actuations during normal plant maneuvers. These SRVs, together with the reactor scram function, provide overpressure protection.

The licensee performed limiting ASME code overpressure analyses based on 102-percent of the EPU power level, the results of which will cause the current SRV set points and upper tolerance limits to change. The licensee's assessment indicates that the SRVs will have sufficient capacity to handle the increased steam flow associated with the operation at the EPU power level.

The design pressure of the reactor vessel and RCPB remains at 1250 psig. The ASME Code allowable peak pressure for the reactor vessel and the RCPB is 1375 psig (110-percent of the design pressure of 1250 psig), which is the acceptance limit for pressurization events. The most limiting pressurization transient is analyzed on a cycle specific basis and this approach would be applicable for each reload cycle. Section 5.5.1.4 and Appendix E of ELTR1 states that the limiting pressurization transients events are the MSIV closure and turbine trip with turbine bypass failure. However, MSIV closure has been determined generically to be the more limiting event. The licensee analyzed MSIV closure event based on an initial dome pressure of 1055 psig with one SRV out of service (OOS), at 102-percent of the EPU RTP. The MSIV position signal scram was assumed to fail and the high-flux signal scram was assumed to shut down the reactor. As provided in Unit 1 SRLR for Cycle 7, the MSIV closure event resulted in a maximum reactor dome pressure of 1301 psig, which corresponds to vessel bottom head pressure of 1331 psig. Therefore, the peak calculated vessel pressure (1331 psig) remains below the ASME limit of 1375 psig.

The licensee used the NRC staff-approved evaluation model ODYN with the equilibrium core to perform the EPU overpressure protection analysis consistent with the generic analysis in Section 3.8 of ELTR2. For the Unit 1 overpressure analysis with Cycle 7 core for EPU operation presented in the SRLR, the maximum calculated pressure meets the ASME code. In addition, the most limiting pressurization transient is analyzed for each reload cycle. Therefore, the NRC staff found that the licensee has demonstrated a conservative analysis of the plant response to overpressure conditions, and determined that no plant modifications are necessary. This provides a reasonable assurance that the probability of gross rupture of RCPB or significant leakage throughout its design lifetime will continue to be exceedingly low. Since the operating ranges of RPV pressure and temperature at the power uprate conditions remain unchanged, its affect on the RCPB design requirement to behave in a nonbrittle manner to minimize rapidly propagating failures is unaffected.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed power uprate on the overpressure protection capability of the plant during power operation.

The NRC staff concludes that the licensee has (1) adequately accounted for the effects of the proposed power uprate on pressurization events and overpressure protection features; and (2) demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, the NRC staff concludes that the overpressure protection features will continue to meet draft GDC-9, 33, 34, and 35 following implementation of the proposed power uprate and, therefore, is acceptable to the NRC staff.

2.8.4.3 Reactor Core Isolation Cooling System

Regulatory Evaluation

The RCIC system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main FW system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with an SBO. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool. The NRC staff's review covered the effect of the proposed power uprate on the functional capability of the system. The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of an LOCA; (2) draft GDC-37, insofar as it requires that ESFs be provided to back up the safety provided by the core design, the RCPB, and their protective systems; (3) draft GDC-51 and 57, insofar as they require that piping systems penetrating containment be designed with appropriate features as necessary to protect from an accidental rupture outside containment and the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.63, insofar as it requires that the plant withstand and recover from an SBO of a specified duration. Specific review criteria are contained in SRP Section 5.4.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The RCIC system of Unit 1 is described in Section 4.7 of the UFSAR. The Unit 1 RCIC system provides core cooling in the event of a transient where the RPV is isolated from the main condenser concurrent with the loss of FW (LOFW) flow, and the RPV pressure is greater than the maximum discharge pressure of the low-pressure core cooling system.

The RCIC system is required to maintain sufficient water inventory in the reactor to permit adequate core cooling following a reactor vessel isolation event accompanied by LOFW. The system design injection rate must be sufficient for compliance with the system limiting criteria to maintain the reactor water level above Top of Active Fuel (TAF). The RCIC system is designed to pump water into the reactor vessel over a wide range of operating pressures. The results of the licensee's evaluation indicate adequate water level margin above TAF, thus, the RCIC injection rate is adequate to meet this design basis event.

An operational requirement is that the RCIC system can restore the reactor water level while avoiding automatic depression system (ADS) timer initiation and MSIV isolation functions associated with the low-low-low reactor water level setpoint (L1). This requirement is intended to avoid unnecessary initiations of these safety systems. The results of the licensee's evaluation indicate that the RCIC system is capable of maintaining the water level outside the shroud above nominal L1 setpoint throughout a limiting LOFW event. Thus, the RCIC injection rate is adequate to meet the requirements for inventory makeup.

Operation of the RCIC system at power uprate conditions did not have any effect on the availability or the reliability of the system, and did not invalidate any of the original design pressures or temperatures for the system components. The RCIC surveillance test range pressure is (in part) based on the maximum normal reactor dome pressure. Because the maximum normal reactor dome pressure increased by 30 psi, the RCIC surveillance test range also was increased by 30 psi.

The licensee further stated that there were no physical changes to the pump suction configuration, and no changes to the system flow rate or minimum atmospheric pressure in the suppression chamber or CST. Additionally, power uprate did not affect the capability to transfer the RCIC pump suction on high suppression pool level or low CST level from its normal alignment, the CST, to the suppression pool, and did not change the existing requirements for the transfer.

Because the licensee has analyzed the LOFW transient for power uprate operation, consistent with the guidelines in Section 4.2 of ELTR2, has conservatively evaluated the pressure performance requirements of the Unit 1 RCIC system, and no RCIC system power dependent functions or operating requirements (flows, pressure, temperature, and NPSH) are added or changed from the original design or licensing bases, the NRC staff finds that the RCIC will continue to meet the NRC's acceptance criteria as delineated in the Regulatory Evaluation section above.

Conclusion

The NRC staff has reviewed the licensee's generic and plant cycle-specific analyses related to the effects of the proposed power uprate on the ability of the RCIC system to provide decay heat removal following an isolation of main FW event and the ability of the system to provide makeup to the core following a small break in the RCPB. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed Power uprate on these events and demonstrated that the RCIC system will continue to provide sufficient decay heat removal and makeup for these events following implementation of the proposed power uprate. Based on this, the NRC staff concludes that the RCIC system will continue to meet the provisions of draft GDC-37, 40, 42, 51, and 57, and 10 CFR 50.63 following implementation of the proposed power uprate. Therefore, the NRC staff finds the RCIC system acceptable for operation at uprated conditions.

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covered the effect of the proposed power uprate on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that ESFs be protected against dynamic effects; (2) draft

GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing; and (3) draft GDC-6, insofar as it requires that decay heat removal systems shall be provided for all expected conditions of normal operation. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The RHR system is described in Section 4.8 of the UFSAR. The RHR system is designed to (1) restore and maintain the reactor coolant inventory and to remove sensible and decay heat from the reactor coolant system and containment following reactor shutdown for both normal shutdown and post-accident conditions. The RHR system is designed to operate in the LPCI mode, shutdown cooling (SDC) mode, suppression pool cooling (SPC) mode, containment spray cooling mode, supplemental SFP cooling mode, and standby cooling/crossties mode. The LPCI mode, as it relates to the LOCA response, is discussed in Section 2.8.5 of this SE. The effects of the power uprate on the other modes are described below.

The uprate results in an increase in decay heat, due to the higher operating power, and the increased amount of heat discharged into the primary containment during an LOCA. This added heat extends the SDC time; however, the RHR pumps and heat exchangers remain capable of maintaining adequate SDC.

During normal plant operation, the SPC function is to maintain the suppression pool temperature below the TS limit. Following abnormal events, the SPC function controls the long-term suppression pool temperature such that the maximum operating temperature limit is not exceeded. The proposed power uprate would increase the reactor decay heat, which increases the heat input to the suppression pool during an LOCA, and results in a higher peak suppression pool temperature. The effect of the proposed power uprate on the suppression pool after a design basis LOCA is discussed in Section 2.6.

The containment spray cooling mode provides suppression pool water to the spray headers in the containment to reduce containment pressure and temperature during post-accident conditions. The effect of the containment spray on containment is discussed in Section 2.6 of this SE.

Supplemental SFP cooling assist Mode uses the RHR heat removal capacity to provide supplemental fuel pool cooling in the event that the fuel pool heat load exceeds the heat removal capacity of the fuel pool cooling and cleanup system. This mode can be operated separately or along with the fuel pool cooling and cleanup system to maintain the fuel pool temperature within acceptable limits.

Standby Cooling/Crossties utilizes the standby coolant supply connection and the RHR crossties to provide additional long-term redundancy to the emergency core and containment cooling systems. This function is not affected by power uprate because the performance requirements for the emergency core and containment cooling systems were not changed.

Based on the NRC staff's review of the licensee's evaluation and rationale, the NRC staff finds that plant operation at the proposed power uprate level will have an insignificant impact on the

SDC mode of the RHR system discussed above, and therefore, no modifications are necessary.

Conclusion

The NRC staff has reviewed the licensee's plant-specific evaluation related to the effects of the proposed power uprate on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the RHR system will continue to meet the provisions of draft GDC-40 and 42 following implementation of the proposed power uprate. Therefore, the NRC staff finds the RHR system acceptable for operation at uprated conditions.

2.8.4.5 Standby Liquid Control System

Regulatory Evaluation

The SLCS provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to effect shutdown. The NRC staff's review covered the effect of the proposed power uprate on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems, preferably of different design principles, be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits; (2) draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and (3) 10 CFR 50.62(c)(4), insofar as it requires that the SLCS be capable of reliably injecting a borated water solution into the RPV at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control. Specific review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Standby Liquid Control System is described in Section 3.8 of the UFSAR. The licensee evaluated the effect of the power uprate on the SLCS injection and shutdown capability. The SLCS is a manually operated system that pumps concentrated sodium pentaborate solution into the vessel in order to provide neutron absorption and is capable of bringing the reactor to a subcritical shutdown condition from RTP.

The licensee stated that an increase in the core thermal power does not by itself directly affect the ability of the SLCS boron solution to bring the reactor subcritical and to maintain the reactor in a safe-shutdown condition. A higher fuel batch fraction, a change in fuel enrichment, or a new fuel design affects the required boron shutdown capability. The minimum reactor boron concentration is increased from 660 parts per million (ppm) to 720 ppm due to the core design changes. The minimum quantity of boron specified in the TS Surveillance Requirement

(SR) 3.1.7.4 is accordingly increased from 186 lb to 203 lb. The SLCS shutdown capability is reevaluated for each reload core.

The licensee performed a plant-specific ATWS analysis with an equilibrium core for EPU. The peak calculated vessel pressure during SLCS operation is 1204 psig for the limiting event. This equates to the pump discharge pressure of 1298 psig for the limiting ATWS case. The pump discharge pressure of 1325 psig given in the TS SR 3.1.7.6 bounds this value.

According to the licensee, the minimum SLCS pump relief valve nominal setpoint for EPU is 1425 psig. A minimum margin of 97 psi has been determined to provide a reasonable assurance against inadvertent relief valve lifting. This margin includes allowance for relief valve setpoint drift and SLCS pump pressure pulsations. The actual margin for EPU operation is 127 psi. The SLCS relief valves are not expected to lift during the ATWS events. The relief valves are periodically tested to maintain this tolerance. Section 50.62(c)(4) requires that each BWR must have a SLCS with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight-percent sodium pentaborate solution. The Unit 1 SLCS meets the ATWS rule requirements of 10 CFR 50.62(c)(4) by meeting the 86 gpm equivalency requirement using the following relationship:

$$(Q/86) \times (M251/M) \times C/13) \times (E/19.8) \geq 1$$

where: Q= expected SLCS flow rate (gpm)
M= mass of water in the reactor vessel and recirculation system at hot rated condition in lb
C= sodium pentaborate solution concentration (weight-percent)
E= Boron-10 isotope enrichment (19.8-percent of natural boron)
M251= mass of water at hot rated condition in a standard BWR/4 251-in. diameter reactor vessel (lb)=628,300 lb

As the changes proposed to TS SR sections 3.1.7.5 and 3.1.7.7 meets the requirements of 10 CFR 50.62, the proposed revisions to these TS sections are therefore acceptable. The NRC staff finds that the SLCS remains consistent with draft GDC 27 and 28. This is due to the SLCS remaining one of two systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed power uprate on the SLCS and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system and demonstrated that the system will continue to provide the function of reactivity control independent of the control rod system following implementation of the proposed power uprate. The present design with the increased quantity of Boron-10 satisfies the draft GDC under which the plant was licensed. The system design will continue to meet draft GDC and current licensing bases in this technical area. Based on this, the NRC staff concludes that the SLCS will continue to meet the provisions of draft GDC-27, 28, and 29, and 10 CFR 50.62(c)(4) following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the SLCS.

2.8.5 Accident and Transient Analyses

AOOs are abnormal transients which are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. DBAs are not expected to occur but are postulated to occur because their consequences would include the potential for the release of significant amounts of radioactive material. They are analyzed to determine the extent of fuel damage expected and to assure that the radiological dose is maintained within 10 CFR 100 guidelines. The applicable acceptance criteria for DBA such as LOCA are based on 10 CFR Part 50.46, 10 CFR Part 50, Appendix K and Draft GDC-40,42 and 44.

The SRP provides further guidelines that (1) pressure in the reactor coolant and MS system should be maintained below 110-percent of the design values according to the ASME Code, Section III, Article NB-7000, Overpressure Protection; (2) fuel cladding integrity should be maintained to ensure that SAFDLs are not exceeded during normal operating conditions and AOOs; (3) an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and (4) an incident of moderate frequency, in combination with any single active component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel cladding perforations are acceptable.

The UFSAR Sections 14.5 and 14.6 evaluate a wide range of potential transients and accidents. Chapter 14 of the UFSAR contains the design basis analyses that evaluate the effects of an AOO resulting from changes in system parameters such as: (1) a decrease in core coolant temperature, (2) an increase in reactor pressure, (3) a decrease in reactor coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, and (6) a decrease in reactor coolant inventory. The plant's responses to the most limiting transients are analyzed each reload cycle and are used to establish the thermal limits. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

Appendix E of ELTR1 identified the set of limiting transients to be considered in each event category. The licensee evaluated the following transients for Cycle 7 as provided in the SRLR:

LFWH (loss-of-feedwater heating)
FWCF (feedwater controller failure)
LRNBP (load rejection, no bypass)
TTNBP (turbine trip, no bypass)
MSIVF (main steam isolation valve closure with flux scram)
Inadvertent HPCI operation
RWE (rod withdrawal error)

The characteristics of the AOO events that determine the operating limit MCPR do not change significantly when reactor power is increased. This has been demonstrated by the initial and reload core analyses for plants with different power levels and power densities and confirmed by the results of subsequent power uprate evaluations. Therefore, the licensee analyzed only the limiting transients.

The NRC staff finds it is acceptable that the following transients were not analyzed in the SRLR since these are not limiting transients:

- PRDS (pressure regulator downscale failure)
- MSIVD (main steam isolation valve closure-direct scram)
- Shutdown Cooling (RHR) Malfunction
- Reactor Vessel Coolant Inventory Decrease----Loss of Auxiliary Power
- Pressure Regulator Failure Open
- Loss of Forced Reactor Coolant Flow
- Recirculation Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
- Core Coolant Flow Increase, Startup of Idle Recirculation Pump, Recirculation Flow Controller Failure
- Control Rod Drop (CRD) Accident
- Inadvertent Opening of a Pressure Relief Valve

2.8.5.1 Events Resulting in a Reactor Vessel Water Temperature Decrease: Loss of a Feedwater Heater, Inadvertent Pump Start

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the RPS, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the RCPB shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; (3) draft GDC-14 and 15, insofar as they require that the core protection system be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (4) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Transients in this group included LFWH, shutdown cooling (RHR) malfunction, and inadvertent RCIC/HPCI pump start. A FW heater can be lost in at least two ways: (1) if the steam extraction line to the heater is shut, causing the heat supply to the heater to be removed, producing gradual cooling of the FW heater, and (2) a bypass line opens so that the FW flow is bypassed around rather than through the heater. In either case, the reactor vessel receives cooler FW, which produces an increase in core inlet subcooling. Due to the resultant negative reactivity coefficient, an increase in power results. This event was analyzed for the Cycle 7 included in the SRLR. The calculated MCPR is 1.22, which is higher than the SLMCPR and, hence, it is acceptable.

EVENT	DISPOSITION
Loss of Feedwater Heater	Evaluated in SRLR for Cycle 7
Inadvertent starting of HPCI Pump	Evaluated in SRLR for Cycle 7

A reliable RPS is provided for Unit 1. Two independent reactivity control systems: CRD system and SLCS are provided. Systems capability of either of them to make the core subcritical under any conditions is unaffected by power uprate.

Conclusion

The NRC staff's SE endorsing ELTR1/2 requires that staff approved acceptable analytical methods be used for the EPU core analysis. The analyses in the SRLR for Cycle 7 used the NRC staff-approved methods. The NRC staff has reviewed the licensee's generic assessment and the SRLR and concludes that it is consistent with the NRC staff's position described in the ELTR1/2 SEs. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that SAFDLs and RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the provisions of draft GDC-6, 9, 14, 15, 27, and 28 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the events stated.

2.8.5.2 Increase in Reactor Pressure: Load Rejection No Bypass, Turbine Trip No Bypass, Closure of Main Steam Isolation Valve

Regulatory Evaluation

A number of initiating events may result in an unplanned increase in reactor pressure and decrease in heat removal from the core. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the RCPB shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Transients in this group included; turbine trip, no bypass; and MS isolation valve closure. A loss of generator electrical load from high power conditions initiates main turbine control valve fast closure. Turbine control valve closure is sensed by the RPS, actuating a reactor scram. Load rejection, without bypass was evaluated for Cycle 7 in the SRLR. The results in the SRLR indicated sufficient margin between the calculated MCPR and the SLMCPR and, hence, is acceptable.

A variety of turbine or nuclear system malfunctions can initiate a turbine trip. Once initiated, all of the main turbine stop valves achieve full closure within about 0.1 second. This event is one of the nuclear pressure events and was evaluated in the SRLR for Cycle 7. The MSIV closure event is bounded by the reactor overpressure protection analysis (MSIV closure with high flux scram) and this limiting event was analyzed in the SRLR for Cycle 7. The results indicate a peak calculated vessel pressure of 1331 psig, which is within the acceptance criterion of 1375 psig and, hence, is acceptable.

EVENT	**DISPOSITION**
Load Rejection No Bypass	Evaluated in the SRLR for Cycle 7
Turbine Trip No Bypass	Evaluated in SRLR for Cycle 7
Closure of MSIV	Analyzed in overpressure protection (Section 2.8.4.2)

A reliable RPS is provided for Unit 1. Two independent reactivity control systems: CRD system and SLCS are provided. Systems capability of either of them to make the core subcritical under any conditions is unaffected by power uprate.

Conclusion

The NRC staff's SE for the ELTRs requires that staff-approved analytical methods be used for the core reload analysis. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the NRC staff's understanding described in the ELTR SEs. In addition, the licensee has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under power uprate conditions. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that SAFDLs and RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the provisions of draft GDC-6, 9, 27, and 28 following implementation of the proposed power uprate. Therefore, the NRC staff finds the capability to mitigate events resulting in a reactor vessel water temperature decrease remains acceptable at uprate conditions.

2.8.5.3 Reactor Vessel Coolant Inventory Decrease: Loss of Feedwater Flow

Regulatory Evaluation

A loss of normal FW flow could occur from pump failures, valve malfunctions, or a LOOP. Loss of FW flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from the fuel following a loss of normal FW flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the RCPB shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

FW Control System failures or reactor FW pump trips can lead to partial or complete loss of FW flow. Loss of FW flow results in a situation where the mass of steam leaving the reactor vessel exceeds the mass of water entering the vessel, resulting in a decrease in the coolant inventory available to cool the core. The licensee performed a plant-specific calculation with a representative equilibrium core for an LOFW. The increased decay heat due to power uprate operation results in a lower reactor water level. This analysis assumed failure of the HPCI system and used only the RCIC system to restore the reactor water level. The reactor level is automatically maintained above the TAF without any operator actions. The results of the LOFW analysis show that the minimum water level inside the core shroud remains above the TAF and hence no cladding failure.

Conclusion

The NRC staff's SE for the ELTRs require that staff-approved analytical methods be used for the core reload analysis. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the NRC staff's understanding described in the ELTR SEs. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under power uprate conditions. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that SAFDLs and RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the provisions of draft GDC-6, 9, 27, and 28 following implementation of the proposed power uprate. Therefore, the NRC staff finds the capability to mitigate events resulting in a decrease in reactor vessel inventory remains acceptable at uprate conditions.

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the transient and the transient itself; (2) the initial conditions; (3) the values of reactor parameters used in the analysis; (4) the analytical methods and computer codes used; and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The rod withdrawal error analysis performed in the SRLR validated that an MCPR value of 1.70 will provide sufficient margin for full withdrawal for reactor power conditions less than 90-percent power. Therefore, the NRC staff finds that the analyses of the transients for single error control rod withdrawal from a subcritical or low-power condition have been confirmed, that the analytical methods and input data are reasonably conservative and that SAFDLs will not be exceeded.

Conclusion

The NRC staff's SE for the ELTRs requires that staff-approved analytical methods be used for the core reload analysis. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the NRC staff's understanding described in the ELTR SEs. In addition, the licensee has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under power uprate conditions. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that SAFDLs and RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the provisions of draft GDC-6, 9, 27, and 28 following implementation of the proposed power uprate. Therefore, the NRC staff finds the capability to mitigate events resulting from an uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions remains acceptable at uprate conditions.

2.8.5.4.2 Positive Reactivity Insertion Event: Continuous Rod Withdrawal During Power Range Operation

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the AOO and the description of the event itself; (2) the initial conditions; (3) the values of reactor parameters used in the analysis; (4) the analytical methods and computer codes used; and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The possibilities for single failures of the reactor control system which could result in a movement or malposition of control rods beyond normal limits have been reviewed. The scope of the review has included investigations of possible rod malposition configurations, the course of the resulting transients AOOs or steady-state conditions, and the instrumentation response to the transient AOO or power maldistribution. The methods used to determine the peak fuel rod response, and the input to that analysis, such as power distribution changes, rod reactivities, and reactivity feedback effects due to moderator and fuel temperature changes, have been examined.

While operating in the power range, it is assumed that the reactor operator makes a procedural error and fully withdraws the maximum worth control rod. Due to the positive reactivity insertion, the core average power increases. If the Rod Withdrawal Error is severe enough, the Rod Block Monitor will sound alarms, at which time the operator will take corrective actions. Even for extremely severe conditions (i.e., for highly abnormal control rod patterns, operating conditions, and assuming that the operator ignores all the alarms and warnings and continues to withdraw the control rod), the fuel cladding integrity safety limit (MCPR) and fuel rod mechanical overpower limits will not be exceeded.

This event was analyzed in the SRLR with different Rod Block Monitor set points. The analysis found the lowest calculated MCPR for this event was 1.40, which is above the SLMCPR and, hence, is acceptable.

Conclusion

The NRC staff's SE for the ELTRs requires that staff-approved analytical methods be used for the core reload analysis. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the NRC staff's understanding described in the ELTR SEs. In addition, the licensee has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under power uprate conditions. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that SAFDLs and RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the provisions of draft GDC-6, 9, 27, and 28 following implementation of the proposed power uprate. Therefore, the NRC staff finds the capability to mitigate single failures of the reactor control system which could result in a movement or malposition of control rods beyond normal limits remains acceptable at uprate conditions.

2.8.5.5 Core Coolant Flow Increase: Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory, Feedwater Controller Failure

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covered (1) the sequence of events; (2) the analytical model used for analyses; (3) the values of parameters used in the analytical model; and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-9, insofar as it requires that the RCPB shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

FWCF transient is initiated when the FW flow controller fails to the maximum demand value. This causes a rapid increase in FW flow. The reactor water level increases until high water level (L8) trip is initiated. When L8 trip set point is reached, a high level main turbine trip occurs, the FW pumps are tripped and a reactor scram is initiated as a consequence of the turbine trip. The failure of the FW controller to a maximum demand event is the most limiting of the vessel inventory increase transients and was evaluated in the SRLR for Cycle 7. The results in the SRLR indicated sufficient margin between the calculated MCPR and the SLMCPR and, hence, is acceptable.

Other equipment malfunctions, operator errors, and abnormal occurrences associated with the inadvertent starting of RCIC or HPCI pumps which could result in increases in the reactor coolant inventory are included in Section 2.8.5.1 of this SE.

Conclusion

The NRC staff's SE for the ELTRs requires that staff-approved analytical methods be used for the core reload analysis. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the NRC staff's understanding described in the ELTR SEs. In addition, the licensee has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under power uprate conditions. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that SAFDLs and RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the provisions of draft GDC-6, 9, 27, and 28 following implementation of the proposed power uprate. Therefore, the NRC staff finds the capability to mitigate events which could result in an increase in reactor coolant inventory acceptable at uprate conditions.

2.8.5.6 Decrease in Reactor Coolant Inventory: Emergency Core Cooling System and Loss-of-Coolant Accidents

Regulatory Evaluation

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents. The NRC staff's review covered (1) the licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of peak cladding temperature (PCT), total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protection and ECCS systems; and (7) operator actions. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of an LOCA; (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of an LOCA; and (4) draft GDC-37, 41, and 44, insofar as they require that a system to provide abundant emergency core cooling be provided so that fuel and clad damage that would interfere with the emergency core cooling function will be prevented. Specific review criteria are contained in SRP Sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The ECCS is described in Section 6 of the UFSAR. ECCS components are designed to provide protection in the event of an LOCA due to a rupture of the primary system piping.

Although DBAs are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For an LOCA, 10 CFR 50.46 specifies design acceptance criteria based on (a) the PCT, (b) local cladding oxidation, total hydrogen generation, (c) coolable core geometry, and (d) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system pipe. Assuming a single failure of the ECCS, the LOCA analysis identifies the break sizes that most severely challenge the ECCS systems and the primary containment. The MAPLHGR operating limit is based on the most limiting LOCA analysis, and licensees perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS for Unit 1 includes the HPCI system, the LPCI mode of the RHR, the CS system and the ADS.

High Pressure Coolant Injection (HPCI)

The HPCI system is designed to pump water into the reactor vessel over a wide range of operating pressures. The primary purpose of the HPCI system is to maintain reactor vessel coolant inventory in the event of a small break LOCA that does not immediately depressurize the reactor vessel. In this event, the HPCI system maintains reactor water level and helps depressurize the reactor vessel.

The licensee indicated that the guidance contained in GE SIL 480 has been implemented. HPCI performance was generically evaluated in Section 4.2 of ELTR2 for a reactor operating pressure increase of up to 75 psi. The evaluation included the effects on pump dynamic head, effects on design water flow rate, a review of vendor test curves, and reviews of turbine speed and horsepower requirements. The increase in reactor pressure increases the maximum required pump set operating head. To enable HPCI pump to deliver the design flow rate at the higher pump set discharge head, the maximum pump and turbine rated speed is increased. The licensee further stated that the HPCI surveillance test range pressure is (in part) based on the maximum normal reactor dome pressure. Because the maximum normal reactor dome pressure increases by 30 psi, the HPCI surveillance test range also is increased by 30 psi.

As the pump and the HPCI pump turbine remain within their allowable operating envelopes, the HPCI system is capable of delivering its design injection flow rate, and the HPCI pump turbine has the capacity to develop the required horsepower and speed. Since the licensee's ECCS-LOCA analysis based on the current HPCI capability demonstrate that the system provides adequate core cooling, the NRC staff finds that HPCI will continue to meet the NRC's acceptance criteria.

Core Spray

The CS system is automatically initiated in the event of an LOCA. When operating in conjunction with other ECCS, the CS system is required to provide adequate core cooling for all LOCA events. There is no change in the reactor pressures at which the CS is required. The CS system sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS system is to provide reactor vessel coolant inventory makeup for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. It also provides long-term core cooling in the event of an LOCA. The increase in decay heat due to power uprate could increase the calculated PCT following a postulated LOCA by a small amount.

The licensee stated that the hardware capabilities of CS equipment required to perform the CS injection function do not change. The ECCS performance evaluation demonstrates that the existing CS system performance capability, in conjunction with the other ECCS as required, is adequate to meet the post-LOCA core cooling requirement for power uprate conditions.

Since the licensee's ECCS-LOCA analysis based on the current CS capability demonstrate that the system provides adequate core cooling, the NRC staff finds that the CS will continue to meet the NRC's acceptance criteria.

Low Pressure Coolant Injection (LPCI)

The LPCI mode of the RHR system is automatically initiated in the event of an LOCA. The primary purpose of the LPCI mode is to help maintain reactor vessel coolant inventory for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. The increase in decay heat due to power uprate could increase the calculated PCT following a postulated LOCA by a small amount. The ECCS performance evaluation demonstrates that the existing LPCI mode performance capability, in conjunction with the other ECCS, is adequate to meet the post-LOCA core cooling requirement. As the LPCI operating requirements are not affected by power uprate, and the licensee's ECCS-LOCA analysis based on the current LPCI capability demonstrate that the system provides adequate core cooling, the NRC staff finds that LPCI will continue to meet the NRC's acceptance criteria.

Automatic Depressurization System

The ADS uses SRVs to reduce the reactor pressure following a small break LOCA when it is assumed that the high-pressure injection systems have failed. After a specified delay, the ADS actuates either on low water level plus high drywell pressure or on sustained low water level alone. This allows the CS and LPCI to inject coolant into the reactor vessel. Plant design requires a minimum flow capacity for the SRVs, and that ADS initiates following confirmatory signals and associated time delay(s). The required flow capacity and ability to initiate ADS on appropriate signals are not affected by power uprate. The licensee stated that the ADS initiation logic and ADS valve control are adequate for power uprate conditions.

The uprate does not affect the protection provided for any of the ECCS features (HPCI, CS, LPCI and ADS) against the dynamic effects and missiles that might result from plant equipment failures. In addition the licensee's ECCS-LOCA analysis, demonstrates that the system provides adequate core cooling, therefore, the NRC staff finds that the ADS will continue to meet the NRC's acceptance criteria.

ECCS Performance

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

Staff-approved codes were used for the LOCA analysis. In the power uprate approach, the LOCA analysis description is based on a limited number of break analyses (one large break and a spectrum of breaks for the small break analyses) instead of the complete set of break-spectrum analyses. This is acceptable for the following reasons:

a) The NRC staff evaluations of several requests for power uprate increase and EPU at BWRs have shown that the change of PCT for power uprates is not significant. The maximum increase in the PCT was small, and was well within the acceptance criteria of 10 CFR 50.46. Since there is only a small change in PCT, power uprate has a negligible effect on the adders used to determine the licensing basis PCT;

b) The ECCS performance characteristics and basic break spectrum response are not affected by a power uprate;

c) The limiting break sizes are well known and have been shown not to be a function of reactor power level;

d) The analyses assume the hot bundle continues to operate at the thermal limits (MCPR, MAPLGHR, and LHGR) which are not changed by the power uprate;

e) The PCT for the limiting large-break LOCA is determined primarily by the hot bundle power, which is expected to increase by a small amount with power uprate;

f) The reload evaluation confirms that the MAPLHGR for each fuel type in the specific reload core is bounded by the MAPLHGR used in the ECCS-LOCA performance analysis; and

g) If the plant is MAPLHGR-limited or if the LOCA analysis results are at (or above) the acceptance criteria limits, a detailed plant-specific analysis for the licensing basis PCT will be performed.

A limited set of analyses need to be performed to determine the impact of power uprate. Because the approach has only a small effect on PCT, the limiting single failure should not change for power uprate conditions in a plant. The LOCA analysis builds on the existing SAFER/GESTR LOCA analyses for a plant. The NRC staff evaluations of past power uprates at BWRs have shown that the basic break spectrum is not affected by power uprate and is expected to have a small effect on the licensing basis PCT. The licensing basis PCT is based on the Appendix K PCT and the effect of power uprate will be based on the delta PCT change from the large break and small break evaluation such that the licensing basis PCT is maximized. Use of the most limiting of the nominal or Appendix K PCT changes for the licensing basis PCT will ensure continued compliance with the requirements for the SAFER/GESTR LOCA application methodology as approved by the NRC.

The power uprate effect on PCT for small recirculation line breaks was larger than the effect on PCT for large line breaks. The increased decay heat associated results in a longer ADS blowdown time leading to a later ECCS system injection and a higher PCT for the small break LOCA. As a result, the limiting LOCA case that defines the PCT at EPU for GE-14 fuel is a small recirculation discharge line break with battery failure.

The PCT was determined based on the calculated Appendix K PCT at rated core flow with an adder to account for uncertainties. As described in a letter dated November 7, 2006, the GE-13 Licensing Basis PCT for the 105-percent OLTP conditions is 1845 degrees F at rated core flow. The GE-14 Licensing Basis PCT for the 105-percent OLTP conditions is 1760 degrees F at rated core flow. At power uprate conditions, the limiting break size is the large break for GE-13, and the 0.06 ft^2 small recirculation line break for GE-14. The changes in PCT are considered small when compared to the PCT margin to the 10 CFR 50.46 licensing limit of 2200 degrees F.

Based on the licensee's plant cycle-specific LOCA analysis, and because the licensee will perform plant cycle-specific evaluations of ECCS-LOCA performance for each fuel reload using approved methods, the NRC staff finds that the ECCS-LOCA performance complies with 10 CFR 50.46 and Appendix K requirements.

Conclusion

The NRC staff's SE for the ELTRs requires that staff-approved analytical methods be used for the core reload analysis. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the NRC staff's understanding described in the ELTR SEs. In addition, the licensee has performed plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under power uprate conditions. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that SAFDLs and RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the provisions of draft GDC-37, 40, 41, 42, and 44, and 10 CFR 50.46 following implementation of the proposed power uprate. Therefore, the NRC staff finds the capability to mitigate events where the inadvertent opening of a safety/relief valve causes a decrease in reactor coolant inventory remains acceptable at uprate conditions.

2.8.5.7 Anticipated Transients Without Scrams

Regulatory Evaluation

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in draft GDC-14 and 15. The regulation at 10 CFR 50.62 requires that:

- each BWR has an ARI system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

- each BWR has an SLCS with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gpm of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel for a given core design.

- each BWR has equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

The NRC staff's review was conducted to ensure that (1) the above requirements are met, (2) sufficient margin is available in the setpoint for the SLCS pump discharge relief valve such that SLCS operability is not affected by the proposed Power uprate, and (3) operator actions specified in the plant's Emergency Operating Procedures are consistent with the generic emergency procedure guidelines/severe accident guidelines (EPGs/SAGs), insofar as they apply to the plant design. In addition, the NRC staff reviewed the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig; (2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200 degree F; (3) the peak suppression pool temperature is less than the design limit; and (4) the peak containment pressure is less than the containment design pressure. The NRC staff also evaluated the potential for thermal-hydraulic instability in conjunction with ATWS events using the methods and criteria approved by the NRC staff. For this analysis, the NRC staff reviewed the limiting event determination, the sequence of events, the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses.

Technical Evaluation

ATWS Prevention and Mitigation Systems are described in Section 9.3.1 of the PUSAR. The ATWS analyses assume that the SLCS will inject within a specified time to bring the reactor subcritical from the hot full power and maintain the reactor subcritical after the reactor has cooled to the cold-shutdown condition. For every reload, the licensee evaluates how plant modifications, reload core designs, changes in fuel design, and other reactor operating changes affect the applicability of the ATWS analysis of record.

The licensee stated that Unit 1 meets the ATWS mitigation requirements defined in 10 CFR 50.62, because (a) an ARI system is installed, (b) the boron injection capability is equivalent to 86 gpm, and (c) an automatic ATWS-Recirculation pump trip (RPT) has been installed. Section L.3 of ELTR1 discusses the ATWS analyses and provides a generic evaluation of the following limiting ATWS events in terms of overpressure and suppression pool cooling: (a) MSIV closure, (b) pressure regulator failure to open, LOOP, and (c) inadvertent opening of a relief valve. The licensee performed a plant-specific ATWS analysis for an equilibrium core to demonstrate that Unit 1 meets the ATWS acceptance criteria. Based on experience, only the limiting cases: MSIV closure and Pressure Regulator Failed Open were analyzed.

The boron injection from SLCS is assumed to start about 2 minutes after the dome pressure reaches ATWS-RPT set point. In addition to boron injection, a number of operator actions, which include lowering water level below the FW sparger and raising water level after hot shutdown boron weight is injected (consistent with the EOIs) are assumed in the Unit 1 ATWS analyses.

Table 9-4 of the PUSAR lists the key input parameters used in the ATWS analyses and Table 9-5 lists the corresponding results (peak vessel bottom pressure, PCT, peak suppression pool temperature, and peak containment pressure). The effects of an ATWS with core instability event occur at natural circulation following a RPT. It is initiated at approximately the same power level as before power uprate, because the maximum extended load line limit analysis upper boundary is not increased. The core design necessary to achieve power uprate operations may affect the susceptibility to coupled thermal-hydraulic/neutronic core oscillations at the natural circulation condition, but would not significantly affect the event progression. Power uprate allows plants to increase their operating thermal power but does not allow an increase in control rod line. Several factors affect the response of an ATWS instability event, including operating power and flow conditions and core design. The limiting ATWS core instability presented in NEDC-24154P-A, Revision 1, Supplement 1, Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors (Supplement 1 - Volume 4), and NEDO-32164, Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS, was performed for an assumed plant initially operating at OLTP and the maximum extended load line limit analysis minimum flow point.

NEDO-32164 indicates that for an unmitigated case, a small fraction of the core experiences locally high peak clad temperature (dryout) and some fuel damage cannot be precluded. For the mitigated case (reduction of reactor water level to reduce core inlet subcooling and direct injection of boron in the presence of power oscillation) extended dryout was not expected. The void reactivity coefficient, fuel response time (fuel rod diameter), and pressure loss coefficients are the parameters important to determining the reactor stability. It also indicates that initial operating conditions of FWH out of service (FWHOOS) and FFWTR do not significantly affect the ATWS instability response. The limiting ATWS evaluation assumes that all FW heating is lost during the event and injected FW temperature approaches the lowest achievable main condenser hot well temperature. The minimum condenser hot well temperature is not affected by FWHOOS or FFWTR. Therefore, the power oscillation for FWHOOS or FFWTR is expected to be no worse than for the normal temperature condition because of a small temperature difference between the initial and final FW temperature.

An on-site audit on the LTSS implementation and ATWS instability of BFN Units 1, 2 and 3 was performed by the NRC staff and its consultant from Oak Ridge National Laboratory on August 8, 2006. The NRC staff's findings based on the on-site audit and the responses to the NRC staff requests are given as follows:

- BFN has implemented the Emergency Procedure Guidelines and Severe Accident Guidelines (EPG/SAG) Revision 1, issued in 1997. The NRC staff reviewed the plant-specific EOIs for ATWS procedures. These instructions were then used in the plant simulator for a demonstration. These EOIs are adequate to manage ATWS events and to mitigate the consequences of unstable oscillations should they occur during these events.

- The effect of power uprate operation on the EOIs is captured automatically by the existing procedures because TVA already recalculates EOI-related parameters on a cycle-specific basis. EOIs and ATWS management strategies need not be modified for power uprate operation.

- Unit 1 has 30-percent turbine bypass capability and SLCS injects through the stand pipes in the lower plenum. The FW spargers are located at approximately the -26 in. level, and TAF is located at -162 in.

- Because the FW pumps are 100-percent steam driven, the only sources of high-pressure injection during an isolation ATWS are HPCI and RCIC. HPCI is used to set a coarse injection flow, and RCIC is used for fine control of the water level.

- The EOIs contain the EPG/SAG recommendations to mitigate ATWS/Stability: (1) prompt water-level reduction to 2 ft below the spargers (~ level -50 in.), and (2) immediate boron injection if power oscillation greater than 25-percent in amplitude is observed. These ATWS mitigation actions will not be modified for power uprate conditions, and should remain effective.

- The EOIs instruct the operator to control water level between the minimum steam cooling-water level (~ -180 in.) and 2 ft below the spargers (~ -50 in.). Operators are trained to maintain level between -100 in. and -60 in.

- The EOIs contain numerous cycle-specific parameters. These parameters are recalculated automatically based on the loading pattern using a computer program. The instructions and their associated flow charts are printed on a cycle-specific basis. Operators are trained using those cycle-specific values as part of the restart training program. Examples of these cycle-specific parameters are: (1) the minimum steam cooling-water level, (2) the hot- and cold-shutdown boron weight, and (3) the suppression pool heat-capacity temperature limit.

- Numerous ATWS transients were performed in the plant simulator during the NRC staff audit. The operators followed the EOIs, and the plant was brought to a safe shutdown for all transient without requiring emergency depressurization. Those transients include: (1) one recirculation pump trip, followed by unstable oscillations and partial failure to scram after the OPRM trip; (2) containment isolation followed by 100-percent failure to insert rods, including complete failure of the ARI; and (3) containment isolation case (2) but SLC was prevented from initiating. These scenarios were performed at the current Unit 2/3 rated power of 105-percent of original rated power. At the request of the auditors, on August 11, 2006, comparative simulator runs were made for a limiting ATWS case at 105-percent thermal power and at 120-percent of original rated power. Although the EPU case was nominally more severe in terms as peak reactor pressure and torus temperature, the EOIs executed in the same manner for 105-percent case and the EPU case, and the operator response to the scenarios likewise executed in the same manner.

- The BFN operators promptly and effectively followed the EOIs as instructed.

As the results of the ATWS analyses meet the ATWS acceptance criteria, the NRC staff finds the plant’s response to an ATWS event is acceptable.

Conclusion

The NRC staff has reviewed the information submitted by the licensee related to ATWS and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on ATWS. The NRC staff has reviewed the licensee’s plant-specific ATWS analyses with an equilibrium core. In addition, the licensee has performed plant-specific reload analyses to confirm that PCT, suppression pool temperature, containment pressure and RCPB pressure limits will not be exceeded during the planned cycle. The NRC staff concludes that the licensee confirmed that ARI, SLCS, and RPT systems have been installed and that they will continue to meet the hardware requirements of 10 CFR 50.62. Therefore, the NRC staff finds the proposed Power uprate acceptable with respect to ATWS.

2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling needs. The NRC staff's review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focused on the effect of changes in fuel design on the analyses for the new fuel storage facilities. The NRC's acceptance criteria are based on draft GDC-66, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The licensee states that both new and spent fuel is stored in the spent fuel storage pools. Each unit has its own fuel pool, which is located on the uppermost level of the reactor building. In this location, the fuel pools are shielded from plant equipment failures that could result in dynamic effects (for example, pipe whips) and from missiles that could result from plant equipment failures (for example, FW pump missiles) by geometric separation, physical barriers, and compartmentalization of operating equipment.

Due to the size and energy associated with the main turbines, the licensee performed additional analyses for power uprate. The three main turbines are separately housed in an adjacent building with about 100 ft of spatial separation between the closest horizontal planes of the fuel pool and the turbine location. All three turbines are laid out in parallel and rotate on an axis perpendicular to the reactor building. This means that the trajectory of any postulated turbine missiles would be square to the reactor buildings (and thus not toward the fuel pools). This orientation results in the main turbines being categorized as a "favorable" orientation with regard to turbine missile failure analyses. A summary discussion of turbine missile probability criteria is provided in Section 11.2 of the UFSAR. Main turbine failure probability analyses were re-performed by the licensee to confirm that the criteria in Section11.2 of the UFSAR would continue to be satisfied considering the turbine modifications that were required for operation at uprated conditions. These analyses confirmed that the licensing bases described in Section 11.2 of the UFSAR continue to be satisfied for power uprate configurations and that the probability of turbine missiles remained acceptably low. As such, the SFP is considered a geometrically safe configuration with regard to main turbine missiles.

The licensee further stated that new and spent fuel is stored in the spent fuel storage pool, and in accordance with TS 4.3.1.1, must maintain a subcritical multiplication factor (k_{eff}) of less than 0.95 when flooded with nonborated water. Furthermore, a reload-specific evaluation is performed to verify that the specific bundle designs being loaded remain bounded by the criticality analysis as discussed in Section 3.5 of the GESTAR.

Based on the NRC staff's review of the licensee's evaluation and rationale, the NRC staff concurs with the licensee that plant operation at power uprate conditions will have an insignificant impact on the new fuel storage discussed above, and therefore, no modifications are necessary. Since, it is not necessary to add or change from the original design or licensing bases, the NRC staff finds that the new fuel storage will continue to meet the NRC's acceptance criteria.

Conclusion

The NRC staff has reviewed the licensee's generic and plant-specific assessment related to the effect of the new fuel on the analyses for the new fuel storage facilities and concluded that the new fuel storage facilities will continue to meet the provisions of draft GDC-66 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the new fuel storage.

2.8.6.2 Spent Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the SFP and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The NRC staff's review covered the effect of the proposed power uprate on the criticality analysis (e.g., reactivity of the spent fuel storage array and boraflex degradation or neutron poison efficacy). The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of an LOCA; and (2) draft GDC-66, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

As discussed in Section 2.8.6.1, the licensee stated that both new and spent fuel is stored in the spent fuel storage pools. Each unit has its own fuel pool, which is located in the reactor building. In this location, the fuel pools are shielded from plant equipment failures that could result in dynamic effects (for example, pipe whips) and from missiles that could result from plant equipment failures (for example, FW pump missiles) by geometric separation, physical barriers, and compartmentalization of operating equipment.

Due to the size and energy associated with the main turbines, the licensee performed additional analyses, as discussed in Section 2.8.6.1. These analyses confirmed that the licensing bases described in Section 11.2 of the UFSAR continue to be satisfied and that the probability of turbine missiles remained acceptably low. As such, the SFP is considered a geometrically safe configuration with regard to main turbine missiles.

The licensee further stated that new and spent fuel is stored in the spent fuel storage pool, and in accordance with TS 4.3.1.1, must maintain a subcritical multiplication factor (k_{eff}) of less than 0.95 when flooded with nonborated water. A spent fuel storage pool criticality analysis was performed by the licensee that confirms that this requirement is met for GE-14 and earlier fuel designs. This analysis applies to all three units which have the same high density storage rack configuration, as detailed in Section 10.3 of the UFSAR. Furthermore, a reload-specific evaluation is performed to verify that the specific bundle designs being loaded remain bounded by the criticality analysis as discussed in section 3.5 of the GESTAR.

Based on the NRC staff's review of the licensee's evaluation and rationale, the NRC staff finds that the proposed power uprate level will have an insignificant impact on the spent fuel storage discussed above, and therefore, no modifications are necessary. Since, it is not necessary to add or change from the original design or licensing bases, the NRC staff accepts the licensee's assessment that the spent fuel storage will continue to meet the NRC's acceptance criteria.

Conclusion

The NRC staff has reviewed the licensee's generic and plant-specific assessment related to the effects of the proposed power uprate on the spent fuel storage capability and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the spent fuel criticality analyses. In addition, the licensee will perform plant-specific reload analyses to confirm that the SFP design will continue to ensure an acceptable degree of subcriticality following implementation of the proposed power uprate. The licensee also performed analyses to confirm that the SFP is considered a geometrically safe configuration with regard to main turbine missiles. Based on this, the NRC staff concludes that the spent fuel storage facilities will continue to meet the provisions of draft GDC-40, 42, and 66 following implementation of the proposed power uprate, and is acceptable to the NRC staff.

2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The NRC staff's review included the parameters used to determine (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all radionuclides other than fission products in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the plant's UFSAR related to LWMSs and GWMSs. The NRC's acceptance criteria for source terms are based on (1) 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the ALARA criterion; and (3) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 11.1.

Technical Evaluation

The core isotopic inventory is a function of the core power level, while the reactor coolant isotopic activity concentration is a function of the core power level, leakage from the fuel, radioactive decay and removal by coolant purification systems. TVA previously submitted a separate license amendment request to implement an AST in accordance with 10 CFR 50.67, which was approved on September 27, 2004, as Amendment 251 to Facility Operating License No. DPR-33 for Unit 1.

TVA discussed radiation sources in the reactor coolant in Section 8.4 of the PUSAR. Radiation sources in the reactor coolant include activation products, activated corrosion products and fission products. TVA used the guidelines in the ELTR1, Section 5.4 to inform its evaluation of the reactor coolant and source terms.

During reactor operation, some stable isotopes in the coolant passing through the core become radioactive (activated) as a result of nuclear reactions. For example, the nonradioactive isotope oxygen-16 is activated to become radioactive nitrogen-16 (N-16) by a neutron-proton reaction as it passes through the neutron-rich core at power. Coolant activation, especially N-16 activity, is the dominant source of radiation in the turbine building and in the lower regions of the drywell. The increase in activation of the water in the core region is in approximate proportion to the increase in the thermal power. TVA's evaluation shows that the activation products in the steam from operation are bounded by the existing design basis concentration. The NRC staff finds that the licensee's evaluation is in accordance with the current licensing basis and follows the guidelines in ELTR1 and SRP Section 11.1, and is acceptable.

Activated corrosion products are the result of metallic corrosion products contained in the coolant water being activated by nuclear reactions as they pass through the core region. Under power uprate conditions, both the FW flow and the activation rate in reactor region increase with power. This results in an increase in activated corrosion product production. TVA calculated that the total activated corrosion product activity is approximately 3-percent higher than the original design basis activity as a consequence of the EPU. The increase in the corrosion product activity is proportional to the increase in reactor power. Therefore, the analysis performed for the EPU is bounding for the 105-percent power uprate.

Fission products in the reactor coolant are present in the steam and in the reactor water as a result of releases from the fuel rods. The activity in the steam is also the noble gas offgas that is included in the design. Using the current licensing basis methodology, TVA calculated offgas rates after 30-minutes decay that are well below the original design basis of 0.35 curie per second. Therefore, the NRC staff agrees with the licensee that the current design basis for offgas activity remains bounding. The increase in the offgas activity is proportional to the increase in reactor power. Therefore, the analysis performed for the EPU is bounding for the 105-percent power uprate.

TVA calculated fission product activity levels in the reactor water that are higher for the EPU than for the current calculated values, using the same methodology. The calculated fission product values increased less than or equal to 13-percent over the current values. The coolant fission product activity levels remain at less than 2-percent of the currently assumed design basis fission product activity. The increase in the coolant fission product activity levels are proportional to the increase in reactor power. Therefore, the analysis performed for the EPU is bounding for the 105-percent power uprate.

Even taking into account the calculated increases to corrosion product and fission product activity due to EPU, the sum of the activated corrosion product activity and the fission product activity remains less than 3-percent of the total design basis activity used in the current BFN analyses. The licensee has assessed that the current activated corrosion product activity and fission product activity design bases are acceptable. In a likewise manner, the current design bases would be bounding for the 105-percent power uprate.

Based on the above evaluations and considering that the licensee has used methodologies in the current licensing basis to evaluate the impact of the uprate on the radiation sources in the reactor coolant, the NRC staff finds the licensee's evaluation acceptable.

Conclusion

The NRC staff has reviewed the radioactive source term associated with the proposed uprate and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. As discussed above, the NRC staff finds that the radioactive source term for the EPU is bounding for the proposed 105-percent power uprate for Unit 1. The NRC staff further concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, and 10 CFR Part 50, Appendix I. Therefore, the NRC staff finds the proposed 105-percent power uprate acceptable with respect to source terms.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

Regulatory Evaluation

The NRC staff reviewed the DBA radiological consequences analyses. The radiological consequences analyses reviewed are the LOCA, FHA,CRD accident, and MSLB. The NRC staff's review for each accident analysis included (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used by the licensee for the calculation of the TEDE. The NRC's acceptance criteria for radiological consequences analyses using an alternative source term are based on (1) 10 CFR 50.67, insofar as it sets standards for radiological consequences of a postulated accident, and (2) GDC-19 in 10 CFR 50 Appendix A, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident. Specific review criteria are contained in SRP Section 15.0.1.

Technical Evaluation

The impact of EPU on the radiological consequences of DBAs is discussed in Section 9.2, of the PUSAR. TVA previously submitted a separate license amendment request to implement an AST in accordance with 10 CFR 50.67, which was approved as Amendment 251 to Facility Operating License No. DPR-33 for Unit 1. The TVA analyses supporting the AST amendments assumed EPU conditions.

In support of the AST amendments, TVA evaluated all significant DBAs currently analyzed for radiological consequences in the UFSAR. These events are the LOCA, MSLB, CRD accident, and FHA. In its previous review for the AST amendments, the NRC staff compared the doses estimated by the licensee to the applicable regulatory criteria and found, with reasonable assurance, that the licensee's estimates of the offsite and control room doses will continue to comply with those regulatory criteria. The SE for the AST amendment stated that the radiological consequences of DBAs would remain bounding up to an RTP of 3952 MWt. Nothing in the EPU submittal or 105-percent power uprate submittal invalidates this previous finding by the NRC staff.

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses performed in support of the proposed uprate and concludes that the licensee has adequately accounted for the effects of operating at thermal power levels as high as the proposed uprate power level. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of postulated DBAs since, as set forth above, the calculated TEDE at the exclusion area boundary at the low-population zone, outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and GDC-19 in 10 CFR 50 Appendix A, as well as applicable acceptance criteria denoted in SRP Section 15.0.1. Therefore, the NRC staff finds the licensee's proposed 105-percent power uprate acceptable with respect to the radiological consequences of DBAs.

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducted its review in this area to ascertain what overall effects the proposed EPU will have on both occupational and public radiation doses and to determine that the licensee has taken the necessary steps to ensure that any dose increases will be maintained ALARA. The NRC staff's review included an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. The NRC staff evaluated how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considered the effects of the proposed uprate on nitrogen-16 levels in the plant and any effects this increase may have on radiation doses outside the plant and at the site boundary from skyshine. The NRC staff also considered the effects of the proposed uprate on plant effluent levels and any effect this increase may have on radiation doses at the site boundary. The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20, 10 CFR 50.67, and draft GDC-11. Specific review criteria are contained in SRP Sections 12.2, 12.3,12.4, and 12.5, and other guidance provided in Matrix 10 of RS-001.

Technical Evaluation

Source Terms:

In general, the production of radiation and radioactive material (either fission or activation products) in the reactor core is directly dependent on the neutron flux and power level of the reactor. However, due to the physical and chemical properties of the different radioactive materials that reside in the reactor coolant, and the various processes that transport them to locations in the plant outside the reactor, several radiation sources encountered in the BOP are not expected to change in direct proportion to the increased reactor power. The most significant of these are:

1. The concentration of noble gas and other volatile fission products in the MSL will not change. The increased production rate of these materials is offset by the corresponding increase in steam flow. Although the concentration of these materials in the steam line remains constant, the increased steam flow results in an increase in the rate these materials are introduced into the Main Condenser and Off Gas systems.

2. For the very short lived activities, most significantly N-16 (7.13 second half-life) and Carbon-15 (C-15 with a 2.4 second half-life), the decreased transit (and decay) time in the MSL, and the increased mass flow of the steam results in a larger increase in these activities in the major turbine building components. Based on the change in travel time of the steam to reach the midpoint of the moisture separator to low pressure turbine crossover piping, the licensee estimates a 32-percent increase in expected dose rates from CLTP dose rates, and a 42-percent increase from OLTP dose rates.

3. The concentrations of nonvolatile fission products, actinides, and corrosion and wear products in the reactor coolant are expected to increase proportionally with the power increase. However, the 15 - 20-percent increase in steam flow is expected to result in small increases in moisture carryover in the steam, resulting in some increased transport of these activities to the balance of the plant. The increases in moisture carryover are expected to be within the current design margin for moisture carryover. Associated increases in dose rates are also expected to be within the shielding design margins for the condensate, FW, and other affected systems.

Radiation Protection Design Features:

1. Occupational and onsite radiation exposures.

The radiation sources in the core are expected to increase in proportion to the increase in power. This increase, however, is bounded by the existing safety margins of the plant design. Due to the design of the shielding and containment surrounding the reactor vessel, and since the reactor vessel is inaccessible to plant personnel during operation, a 5-percent increase in the radiation sources in the reactor core will have no effect on occupational worker personnel doses during power operations. Similarly, the radiation shielding provided in the BOP is conservatively sized such that the increased source terms discussed above are not expected to significantly increase the dose rates in the normally occupied areas of the plant. Radiation dose rates in “steam-affected” areas of the plant are estimated to increase. These areas (include the reactor and turbine steam tunnels, moisture separator rooms, turbine rooms, high and low pressure heater rooms, condenser rooms, moisture separator drain pump and tank rooms, steam packing exhauster rooms, steam jet air ejector rooms, and hydrogen recombiner rooms) are all currently designated as high-radiation areas and personnel access to them is restricted and controlled accordingly. The existing radiation zoning design (e.g., the maximum designed dose rates for each area of the plant), for areas outside the steam-affected areas, will not change as a result of the increased dose rates.

During testing of each unit, the licensee will perform sampling and measurements to determine the radiochemical quality of the reactor water, FW, and gaseous releases. In addition, general area dose rates will be measured at plant locations susceptible to increased N-16 and neutron doses as a result of the power increase (i.e., walkways and rooms adjacent to steam-affected areas of the plant, FW pump rooms, and the Turbine Building roofs). These measurements and sampling will be performed as each unit reaches 90, 100, and 105 percent of the OLTP.

Operating at a higher power level will result in an increased core inventory of radioactive material that is available for release during postulated accident conditions. The plant shielding design must be sufficient to provide control room habitability, per GDC-19, and operator access to vital areas of the plant, per NUREG 0737 item II.B.2, during the accident. As part of a change to Units 1, 2, and 3 design basis, the licensee recently re-calculated the radiological consequences of the postulated DBAs using the AST in accordance with the provisions in 10 CFR 50.90 and 10 CFR 50.67. The AST provides more realistic assumptions, than the current design basis source term, on the timing and mechanisms of radioactive material release from the core during postulated accident conditions. In re-evaluating the DBAs, the licensee re-calculated the radiation doses associated with control room habitability, and post-accident vital areas access, at the proposed power level of 3952 MWt. The NRC staff documented its review and approval of the licensee’s use of AST in the SE dated September 27, 2004.

Therefore, following implementation of this power uprate, the BFN units will continue to meet their design basis in terms of radiation shielding, in accordance with the criteria in SRP Section 12.4, GDC-19, and NUREG-0737, item II.B.2.

2. Public and offsite radiation exposures.

There are two factors, associated with this power uprate, that may impact public and offsite radiation exposures during plant operations. These are the possible increase in gaseous and liquid effluents released from the site, and the increase in direct radiation exposure from radioactive plant components and solid wastes stored onsite. As described above, this power uprate will result in an increase in gaseous effluents released from the plant during operations. This increase is a minor contribution to the radiation exposure of the public. The nominal annual public dose from plant gaseous effluents for BFN is typically a small fraction of the design criteria of 10 CFR Part 50, Appendix I, and the Environmental Protection Agency’s dose limits in 40 CFR Part 190 (as referenced by 10 CFR 20.1301(e)). For example during the reporting period of January 1 to December 31, 2004, the maximum dose to a member of the public resulting from airborne releases from BFN, was less than 1-percent of the dose criteria in 10 CFR Part 50, Appendix I and 40 CFR 190. Even with the conservative assumption that the resumption of Unit 1 power operations increases this by 50 percent, the dose to the public from airborne effluents will continue to be well below these applicable regulatory requirements.

This power uprate will also result in increased generation of liquid and solid radioactive waste. The increased condensate feed flow associated with the uprate results in faster loading of the condensate demineralizers. Similarly, the higher feed flow introduces more impurities into the reactor resulting in faster loading of the RWCU system demineralizers. Therefore, the demineralizers in both of these systems will require more frequent back washing to maintain them. The licensee has estimated that these more frequent backwashes will increase the volume of liquid waste, that will need processing, by 4 percent and an increase in processed solid radioactive waste by 15 percent. These increases are well within the processing capacity of the radwaste systems and are not expected to noticeably increase the liquid effluents or solid radioactive waste released from the plant. Therefore, these increases will have a negligible impact on occupational or public radiation exposure.

Skyshine is a physical phenomenon associated with gamma radiation that is emitted skyward, during radioactive decay. As this radiation interacts with air molecules, some is scattered back down to the ground where it can expose members of the public. Since there is significantly less radiation shielding above the steam components in the turbine building, than there is to the sides of these components, skyshine from N-16 gammas can be a significant contributor to dose rates outside plant buildings (both onsite and offsite). As discussed above, the licensee has estimated that plant operations at uprated conditions will increase the N-16 and C-15 activity in the turbine building. In addition, the practice of injecting hydrogen into the reactor FW, to reduce stress corrosion cracking, significantly increases the fraction of N-16 in the reactor water that is released into the steam during power operations. The latter effect is somewhat mitigated by pre-treating the system with noble metals. Prior to initiating reduced hydrogen injection chemistry, the licensee performed radiation surveys on-site with Units 2 and 3 operating at OLTP to determine the magnitude of gamma dose rates on-site. Based on these survey results, the licensee calculated that the dose rates at the nearest site boundary (i.e., 3850 ft) was 0.04 micro R per hour per unit. This dose rate equates to a maximum annual dose to an individual (if they resided outside the nearest site boundary) of approximately 1.1 mrem from all three units operating at OLTP. Based on their experience with Units 2 and 3, the licensee has estimated that reduced hydrogen injection chemistry will increase this dose rate by 25 percent. Therefore, adjusting the calculated dose rate by a factor of 1.25 to account for hydrogen injection, and a factor of 1.42 to account for the reduced decay time (from OLTP to 120-percent OLTP) from increased steam flow rate, results in a maximum annual dose to an offsite member of the public of approximately 1.9 mrem. This is well within the annual limit of 25 mrem to an actual member of the public, as referenced by 10 CFR 20.1301(e).

Operational Radiation Protection Programs:

The increased production of nonvolatile fission products, actinides and corrosion and wear products in the reactor coolant may result in proportionally higher plate-out of these materials on the surfaces of, and low flow areas in, reactor systems. The corresponding increase in dose rates associated with these deposited materials will be an additional source of occupational exposure during the repair and maintenance of these systems. However, the current ALARA program practices at Units 1, 2, and 3 (e.g., work planning, source term minimization, etc.), coupled with existing radiation exposure procedural controls, will be able to compensate for the anticipated increases in dose rates. Therefore, the increased radiation sources, as discussed above, will not adversely impact the licensees ability to maintain occupational and public radiation doses resulting from plant operation to with the applicable limits in 10 CFR Part 20 and ALARA.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed uprate on radiation source terms and plant radiation levels. The NRC staff concludes that the licensee has taken the necessary steps to ensure that any increases in radiation doses will be maintained ALARA. The NRC staff further concludes that the proposed uprate meets the requirements of 10 CFR Part 20 and draft GDC-11. Therefore, the NRC staff finds the licensee's proposed uprate acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained as low as reasonably ALARA.

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implement the proposed power uprate. The NRC staff's review covered changes to operator actions, human-system interfaces, procedures, and training needed for the proposed power uprate. The NRC's acceptance criteria for human factors are based on the following documents: draft GDC-11; 10 CFR 50.120; 10 CFR Part 55; ANSI/ANS Standard 58.8 (1994/2001), Time Response Design Criteria for Safety-Related Operator Actions; and the guidance in GL 82-33. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and Chapter 18.0.

Technical Evaluation

The NRC staff has developed a standard set of questions for the review of the human factors area. The licensee has addressed these questions in its application. Following are the NRC staff's questions, the licensee's responses, and the NRC staff's evaluation of the responses.

1. Changes in Emergency and Abnormal Operating Procedures

This section evaluates how the proposed power uprate will change the plant Emergency Operating Procedures (EOPs) and Abnormal Operating Procedures. (SRP Section 13.5.2.1.)

TVA designates EOPs and Abnormal Operating Procedures as Emergency Operating Instructions (EOIs) and Abnormal Operating Instructions (AOIs), respectively. The EOIs are symptom-based. The changes in EOIs and AOIs required for the EPU implementation consist of revisions to previously-defined numerical values within the EOIs and AOIs. These parameter values represent plant status data such as rated reactor thermal power and the heat capacity temperature limit, however, the definition of parameters used in the AOIs and EOIs will not change. The changes do not result in changes in the operating philosophy or the accident mitigation philosophy.

Therefore, because no new procedures or changes to operating or accident management philosophy are required, necessary data changes to EOIs and AOIs will be implemented, and training to address these changes will be provided, the NRC staff finds the licensee's proposed changes in this area to be acceptable.

2. Changes to Operator Actions Sensitive to Power Uprate

This section evaluates any new operator actions needed as a result of the proposed uprate and changes to any current operator actions related to emergency or abnormal operating procedures that will occur as a result of the proposed uprate (SRP Section 18.0).

TVA requests credit be given for a manual action to terminate the drywell coolers within 2 hours of entry into the safe shutdown procedure during an Appendix R event. The manual action of shutting off the drywell coolers can be performed in the control room or in two remote shutdown locations outside of the control room. The licensee has indicated that shutting off the drywell coolers is a simple action that is not extensive. It was indicated that in the analysis, the time available in the uprated conditions is 2 hours and the time required to terminate the drywell coolers is well within this time frame. The action is the same in the remote locations as it is in the control room. The two locations (i.e., switchgear room) are separate. The licensee has validated the timing for the procedures to ensure the manual action can be performed. However, the NRC staff notes that issuance of the completed procedures which require licensee validation of other considerations regarding the feasibility to complete the prescribed operator actions are not complete. This validation is expected to be complete before restart. The NRC staff has determined, that provided the manual action is feasible and does not adversely affect the safety of the plant, the revision the BFN Appendix R fire safe shutdown operating instructions to manually terminate drywell cooling within 2 hours of entry into the procedure is acceptable.

The change in parameter values, including core decay heat and thermal power level, that are associated with power uprate conditions could affect the timing of actions provided in the EOIs and AOIs. In response to the NRC staff's questions, the licensee provided a detailed table including a description of the operator actions, the operator action times available used in the current power analysis, the operator action times available, and the time required for the operator to perform the actions credited in the UFSAR. With the exception discussed below, the proposed power uprate conditions do not affect the steps required for the operators to perform their actions nor do the proposed conditions affect the time it will take the operators to complete the required action.

The only impact on operator action times is a decrease in the time available to complete initiation of the CAD system. The time available for CAD system initiation in the current thermal level power analysis is 42 hours. The time available used in the analysis is 32 hours. When notified to initiate CAD, the operator is required to complete the action in 5 minutes, for both the current power level and in the analysis. The proposed power uprate does not affect the time required to complete the operator actions, nor does the change in the time required for CAD system initiation affect operator action times.

Based on the licensee's description of the actions credited in the UFSAR, the time available for CAD system initiation, and the time required to initiate the CAD system, the NRC staff finds that the operators will be able to successfully accomplish the actions required to support the proposed power uprate.

3. Changes to Control Room Controls, Displays and Alarms

This section evaluates any changes the proposed power uprate will have on the operator interfaces for control room controls, displays, and alarms.

TVA stated that there are no major changes planned for the control room, the control room displays, or control room alarms as a result of the power uprate. As a result of changed process conditions and the installation of new equipment, changes to instrumentation spans, alarm settings, and actuation setpoints are necessary. Setpoints changed include the APRM, flow biased scram, rod block setpoints, turbine stop valve closure, control valve fast closure scram bypass setpoints, and the MSL high flow isolation setpoints.

Control room instrumentation and operator aid changes will be modified in accordance with the BFN plant modification process. Various labels, sketches, and markings posted in the main control room will be modified. Training and implementation requirements, and simulator impact are identified and tracked. Verification of successful completion of operator training is required as part of the modification closure process.

As TVA has identified the necessary control room modifications and training to address the changes to the control room controls, displays and alarms, the NRC staff finds the licensee's proposed modifications and training plan in this area to be acceptable.

4. Changes to the Safety Parameter Display System

This section assesses any changes to the safety parameter display system, and how the operators will know of the changes (SRP Section 18.0).

A safety parameter display system (SPDS) was not in place prior to the extended shutdown of Unit 1. An SPDS similar to the system in Units 2 and 3 is being installed as a part of the modifications required for restart. The design, intent, and information presented on the SPDS are the same as the SPDS for Units 2 and 3. The analog and digital inputs for the SPDS will be reviewed to determine the effects as a result of the power uprate. In a letter dated April 8, 1987, TVA committed to completing the installation of the SPDS prior to startup. The SPDS will be included in control room staff training, and will not affect EOI execution.

As TVA has committed to installing an SPDS and training the control room staff on the SPDS prior to startup, the NRC staff finds these proposed changes acceptable.

5. Changes to the Operator Training Program and the Control Room Simulator

This section evaluates any changes to the operator training program and the plant-referenced control room simulator resulting from the proposed power uprate and the implementation schedule for making the changes (SRP Sections 13.2.1 and 13.2.2).

The licensee indicated that simulator and classroom training will be completed during the last training phase prior to restart. This training should include normal operating procedure actions required to achieve power uprate, power ascension testing, and physical plant changes as modeled in the simulator.

Changes on the simulator will be installed prior to Unit 1 restart to support training of all operating crews. Setpoint changes corresponding to 105-percent OLTP will also be reflected on the simulator prior to completion of training. Acceptance testing of the simulator will be conducted to benchmark its performance and will be implemented in accordance with ANSI/ANS 3.5. The performance of the simulator will be validated against the expected power uprate response and then against operating data collected during power uprate implementation and startup testing. Based on the results of the validation, TVA has indicated that any necessary adjustments to the simulator model will be made.

Based on the above, the NRC staff is satisfied that TVA will develop and implement the necessary licensed operator training and will update the simulator to reflect EPU conditions.

Conclusion

The NRC staff has reviewed the changes to operator actions, human-system interfaces, procedures, and training required for the proposed uprate and concluded that the licensee has (1) appropriately accounted for the effects of the proposed uprate on the available time for operator actions and (2) taken appropriate actions to ensure that operator performance is not adversely affected by the proposed uprate. The NRC staff further concludes that the licensee will continue to meet the provisions of draft GDC-11, 10 CFR Part 50.120, and 10 CFR Part 55 following implementation of the proposed uprate. Therefore, the NRC staff finds the licensee's proposed uprate acceptable with respect to the human factors aspects of the required system changes.

2.12 Power Ascension and Testing Plan

2.12.1 Approach to Uprate Power Level and Test Plan

Regulatory Evaluation

The purpose of the uprate test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed uprate power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at power uprate conditions. The NRC staff's review included an evaluation of (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance; (2) transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level; and (3) the test program's conformance with applicable regulations. The NRC's acceptance criteria for the proposed uprate test program are based on 10 CFR Part 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service. Specific review criteria are contained in SRP Section 14.2.1.

Technical Evaluation

Restart Test Program (RSTP)

Section 13.5 of UFSAR, provides an overview of the initial power ascension test program from initial fuel loading through 100-percent power. The NRC staff reviewed Section 13.5.2.2 of the UFSAR for Unit 1, and Section 13.5.2.3 for Units 2 and 3, which presented a general purpose, description, and acceptance criteria of the initial startup testing. Additional information was reviewed by the NRC staff, which described the startup and power test program performed to demonstrate that the plant is capable of operating safely and satisfactorily.

In the interim uprate request dated September 22, 2006, TVA indicated that to demonstrate that SSCs will perform satisfactorily upon restart and under power uprate conditions the applicable startup test program proposed as part of the June 28, 2004, 120-percent request will be conducted as described in Section 10.4 of the PUSAR and Enclosure 8 to the June 28, 2004, submittal. This program was supplemented by the licensee in letters dated February 23, April 25, and August 25, 2005. The planned power ascension testing at Unit 1 is intended to provide a controlled and systematic testing program for the power levels above current licensed thermal power to power uprate conditions.

Unit 1 will perform a comprehensive restart test program. The RSTP was developed to control testing of safety system safe shutdown functional requirements and is one of many special programs contained within the regulatory framework developed for the restart of Unit 1. The RSTP is similar to that previously approved by the NRC staff for the restart of Units 2 and 3 in 1991 and 1995, respectively. TVA submitted to the NRC a proposed update to the regulatory framework for Unit 1 in 2002, which addressed regulatory requirements, special programs, commitments, and TS changes. In general, TVA intends to restart Unit 1 consistent with the criteria and methods developed previously in support of the restart for Units 2 and 3. As part of the review, the NRC staff performed a programmatic review of the RSTP and TVA's return-to-service program for Unit 1, which includes post-modification, post-maintenance, restart, and surveillance testing. The program will also utilize existing BFN design criteria, TVA programs, and procedures. The RSTP tests include both the pre-operational condition tests and the power ascension tests. The test requirements are identified in various Baseline Test Requirements Documents.

In a letter dated June 15, 2005, the NRC staff requested that TVA provide more specific and detailed information regarding implementation of the RSTP for Unit 1. On August 15, 2005, TVA provided its response, which included a description of the procedures under which the RSTP was developed and controlled including a line-item description of the specific testing planned and a comparison to the testing performed under the RSTPs for both Units 2 and 3. In a letter to the NRC staff dated February 18, 1992, TVA provided a comparison of the Unit 1 planned power ascension testing to the power ascension testing previously approved for Unit 3. The information referenced 31 tests planned as part of the Unit 1 power ascension testing program and is identical to that performed for Unit 3.

Development of the Unit 1 RSTP and identification of the associated test requirements are administratively controlled by Procedure 1-TI-452, Unit 1 Restart Test Program, which specifies the review and approval of the restart test requirements, associated acceptance criteria, and test results. The test requirements are documented in the Baseline Test Requirements Documents, which are developed, reviewed, and approved in accordance with Procedure 1-TI-469, Baseline Test Requirements. Consistent with the Unit 3 RSTP, a multi-disciplinary review group reviews and approves the RSTP test requirements identified for each system as well as the results of the testing for each system. In addition, TVA is also performing system and component post-maintenance, post-modification, calibration, normal surveillance, and power ascension testing, as required, to ensure that systems will operate in accordance with their design requirements. This testing, which is not part of the RSTP, is identified and controlled in accordance with the quality assurance requirements of 10 CFR Part 50, Appendix B.

Restart testing for Unit 1 is controlled in accordance with Procedure 1-TI-453, Unit 1 Startup Test Instruction, which provides coordination, tracking, and control of system testing. Under this procedure, system test specifications are developed to define the minimum testing requirements, their bases, acceptance criteria, and the test procedures that will be used to satisfy the test requirements for selected systems. The system test specifications are intended to encompass all functional testing beyond the scope of static installation and minor post-maintenance testing, and includes component tests, loop calibrations, post-modification functional tests, technical instructions, special tests, and TS surveillance tests. Consistent with the RSTP, TVA is utilizing existing surveillance and testing procedures to the extent possible to perform these tests. Where planned testing is not covered by existing procedures, explicit test instructions are developed to perform the testing.

The NRC staff concludes, through comparison of the documents referenced above and a review of the initial startup and test program described in Section 13.5 of the UFSAR, that the proposed test program adequately identified (1) all initial power ascension tests performed at a power level of equal to or greater than 80 percent of the OLTP level; (2) all initial test program tests performed at power levels lower than 80 percent of the OLTP level that would be invalidated by the uprate; and (3) differences between the proposed power-ascension test program and the portions of the initial test program identified by the previous criteria. The NRC staff also concludes that with respect to the program implementation methodology, including the programmatic and administrative controls described in TVA procedures, and in the enclosures to their letters, the startup and test program is acceptable for operation up to 105-percent OLTP.

Post-Modification Testing Requirements for SSCs Important to Safety Impacted by Plant Modifications

As part of the plant modification control process, TVA reviews all Design Change Notices to identify and document appropriate post-modification testing requirements. Design Change Notices are prepared and controlled in accordance with TVA procedure SPP-9.3, Plant Modifications and Engineering Change Control, which requires that design engineers identify and document any required verification and/or special testing requirements. In accordance with TVA procedure SPP-8.3, Post-Modification Testing, system engineers review the Design Change Notices to confirm that all necessary post-modification testing has been specified and are responsible for reviewing and concurring with acceptability of each test. These programmatic controls ensure that the testing required to ensure system performance requirements following each design change are met and expected system response is confirmed. Testing is identified and performed with acceptable results prior to turnover of the system for operation, which is consistent with industry programs developed to ensure compliance with the quality assurance requirements of 10 CFR 50, Appendix B.

TVA stated that they evaluated the modifications currently planned to support the power uprate and have determined that they do not constitute a material alteration to the plant. Some of the planned modifications considered by the NRC staff in its review of the application are listed in Section 1.4. Many of the modifications for Unit 1 are in support of operation at 120 percent, should the 20-percent uprate be approved during the upcoming operating cycle. These modifications will be in place but not all are required for operation at 105-percent of OLTP. TVA stated that evaluations of the actual test results may identify the need for additional tests or the revision of the tests planned and therefore, the final test plan may be revised. The NRC staff also reviewed TVA's submittal of their aggregate impact of the plant modifications, setpoint adjustments, and parameter changes that could adversely impact the dynamic response of the plant to anticipated initiating events.

Because the NRC staff is relying in part on satisfactory completion of CFS transient testing in determining that the CFS is acceptable for uprated power operation, a License Condition will be established to require the satisfactory completion of the transient tests that are deemed to be necessary. In a letter dated September 27, 2006, the NRC staff informed the licensee that the following License Condition would be established in this regard:

> During the power uprate power ascension test program and prior to exceeding 30 days of plant operation above a nominal 3293 megawatts thermal power level (100-percent OLTP) or within 30 days of satisfactory completion of steam dryer monitoring and testing that is necessary for achieving 105-percent OLTP (whichever is longer), with plant conditions stabilized at 105-percent OLTP, TVA shall trip a condensate booster pump, a condensate pump, and a main feedwater pump on an individual basis (i.e., one at a time). Following each pump trip, TVA shall confirm that plant response to the transient is as expected in accordance with previously established acceptance criteria. Evaluation of the test results for each test shall be completed and all discrepancies resolved in accordance with corrective action program requirements and the provisions of the power ascension test program.

In Table 1 of Enclosure 3 of the June 28, 2004, letter, TVA indicated that pump trip testing as discussed above would be performed for modification activities associated with the condensate pumps, condensate booster pumps, and the FW pumps, and that such testing would be performed as part of the currently planned power ascension test program.

The NRC staff concludes that the testing program proposed by TVA should adequately demonstrate that any related modifications, necessary to support operation at the uprated power level, will be adequately tested. Specifically, the NRC staff concludes that the test program appropriately includes those SSCs (1) impacted by modifications; (2) used to mitigate an AOO described in the plant design basis; and (3) supported a function that relied on integrated operation of multiple systems and components.

Integrated Testing

In the June 28, 2004, request for uprate of Unit 1 to 120-percent OLTP, the licensee requested elimination of two large transient tests (LTTs). These LTTs are an MSIV closure test and a generator load reject test. To determine whether some tests can be omitted, plant design details (such as configuration, modifications, and relative changes in setpoints and parameters), equipment specifications, operating power level, test specifications and methods, operating and emergency operating instructions; and adverse operating experience must be considered and addressed.

As stated previously, TVA intends to restart Unit 1 consistent with the criteria and methods developed previously in support of the restart for Units 2 and 3. Since its shutdown in 1985, Unit 1 has been defueled, systems have not operated and have generally been maintained in a layup condition. As part of the Unit 1 recovery, TVA has indicated that many of the design analyses have been revised in anticipation of approval of power uprate at 120 percent. Many upgrades and significant modifications will be implemented including the replacement of some large- and small-bore piping; installation of cable trays, conduits, supports, and pipe hangers; refurbishment of various large pumps and motors (including the main FW, condensate, condensate booster and recirculation pumps); rewind of the turbine-generator; installation of main bank transformers; and modification of various control room indicators. Upon completion of the recovery, Unit 1 would include a combination of new and refurbished components and new and old piping, supports, cable trays, cables, etc.

An acceptable test program ensures that SSC capabilities to perform these specified/analyzed functions are initially verified with adequate precision and accuracy, that necessary SSC and plant baseline performance data is obtained, that deficiencies are identified and corrected, and that activities are conducted in a sequence and manner which minimize operational reliance on untested SSCs/safety functions. This provides high degrees of assurance of SSC and overall plant readiness for safe operation within the bounds of the design and safety analyses, assurance against unexpected or unanalyzed SSC/plant behavior, and assurance against early SSC/safety function failures in service.

The NRC staff found that the approach presented by the licensee for the RSTP discussed previously is acceptable to address the functionality of the tested systems. However, given that many of the design analyses have not been validated, the extent of the modifications and the lack of actual operating data since installation of these modifications, the response of this unit to significant transients may be different. It is possible that not only will the response be different from when it was tested and started-up in late 1973 and early 1974, but it could be different from the new design analyses conservatively performed to support the proposed 120-percent request.

The NRC staff reviewed the licensee's proposed approach, and the initial transient test program requirements that were specified for original plant licensing. This approach focuses on the use of TS surveillance testing as a substitute for more integrated system testing. While component testing will demonstrate component performance and surveillance testing can be used to partially confirm continued proper performance, this testing is insufficient to demonstrate satisfactory integrated plant performance. Integrated testing is necessary to effectively confirm plant response and analyses at the uprated conditions. Therefore, the NRC staff found the licensee's proposal to eliminate LTTs unacceptable.

In a letter dated June 28, 2006, the NRC proposed a License Condition requiring both LTTs be performed. This License Condition was acknowledged by TVA in a letter dated July 31, 2006, with certain modifications requested for the proposed power level and time frame. Subsequently, in the September 22, 2006, letter, TVA proposes to establish the following commitment for completing two LTTs for the Unit 1:

1. A large transient test that simulates the rejection of generator load will be completed within 30 days of reaching 105-percent OLTP.

2. An MSIV closure with valve position scram large transient test will be performed within 30 days of reaching 115- to 120-percent OLTP.

The NRC staff reviewed the September proposal and found the first commitment acceptable, but found that the licensee's commitment inferred approval for operation above 105-percent OLTP. As no such approval has been made, the NRC staff notified the licensee in a letter dated September 27, 2006, that both LTTs will be performed consistent with the NRC staff's proposal. Therefore, the NRC staff proposes the following License Condition:

> During the power uprate power ascension test program and prior to exceeding 30 days of plant operation above a nominal 3293 megawatts thermal power level (100-percent OLTP) or within 30 days of satisfactory completion of steam dryer monitoring and testing that is necessary in order to achieve 105-percent OLTP (whichever is longer), with plant conditions stabilized at 105-percent OLTP, TVA shall perform a MS isolation valve closure test and a turbine generator load reject test. Following each test, TVA shall confirm that plant response to the transient is as expected in accordance with previously established acceptance criteria. The evaluation of the test results for each test shall be completed, and all discrepancies resolved, prior to resumption of power operation.

<u>Conclusion</u>

The NRC staff has reviewed the power uprate test program, including plans for the initial approach to the proposed maximum licensed thermal power level, transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and the test program's conformance with applicable regulations. The NRC staff concludes that the proposed test program, as supplemented by the two NRC License Conditions provides adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the proposed power uprate, or modified to support the proposed power uprate, will perform satisfactorily in service. Further, The NRC staff finds that there is reasonable assurance that the testing program satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Therefore, the NRC staff finds the proposed test program acceptable for operation at 105-percent OLTP.

2.13 License Renewal

The licensee submitted an application for the renewal of the Unit 1 licensee on December 31, 2003. The LRA included aging management reviews and aging management programs (AMPs) to support continued operation at the 100-percent power level. The 120-percent power uprate application was independently submitted as an amendment request and power uprate conditions were not addressed in the LRA. By letter dated January 31, 2006, the licensee submitted an annual update to report recent changes that materially affect the contents of the LRA. The LRA application was approved by the NRC staff on May 4, 2006.

Because the LRA review applied to the period of extended operation, licensees were not required to have implemented all activities credited for aging management when the renewed licenses were issued. A list of commitments for remaining actions requiring implementation was included in the license renewal FSAR supplement. The renewed license includes a condition requiring completion of these commitments prior to operating beyond the expiration date of the original 40-year license. Licensee implementation of many of these commitments will be verified by regional inspection. However, some of these commitments were required to be complete prior to restart of Unit 1.

The restart of Unit 1 involves plant changes that include both physical modifications and new operating practices to increase power from the current 100-percent licensed power level. Since the LRA was to be based on 100-percent power, the LRA did not necessarily consider plant changes to support either the 105- or 120-percent uprates, including those in progress to support the Unit 1 restart. Therefore, consideration of the modifications and operating practices that potentially result in new materials, aging management reviews or AMPs that could have an impact on licensee renewal is included in this section.

2.13.1 Aging Management Program

Regulatory Evaluation

For license renewal, an aging management review is required by 10 CFR Part 54 to ensure that SSCs have been adequately evaluated for aging effects so that appropriate AMPs are in place to manage the applicable aging effects. Typically, the LRA represents the current configuration of the plant and future changes are included in the annual update. For Unit 1, the plant configuration represented in the LRA was understood to be the configuration at the time of restart which coincided with completion of plant changes to support EPU. Appendix F of the LRA, *Integration of Browns Ferry Unit 1 Restart and License Renewal Activities*, includes an overview of the Unit 1 restart plan and a description of planned modifications to be completed prior to restart. Appendix F also includes a brief summary of the impact of plant changes on licensee renewal showing that there is no impact on any AMP or time-limited aging analysis. As a majority of the modifications that will be installed will be initially operated at 105 percent, the NRC staff has taken the conservative analysis performed at 120 percent, and verified that the AMP is acceptable for operation at 105-percent power uprate conditions.

Technical Evaluation

The NRC staff reviewed the information included in the LRA, including Appendix F, and determined that additional information was required to determine how plant changes that could impact the aging management review would be considered for license renewal. Some of the NRC Staff's review concerning aging management is contained in Section 2.1.2.

To support both the power uprate and the license renewal reviews, the NRC staff requested that the licensee provide a discussion to clarify how the SSC already in scope of the LRA are going to be impacted by the intended modifications and conditions. By letter dated June 15, 2006, TVA clarified that the LRA was developed in parallel with the June 28, 2004, license amendment request. The licensee also clarified that the conditions related to implementation were conservatively reviewed during the license renewal process and at 120-percent uprate conditions were evaluated and incorporated into the LRA. Additionally, the licensee identified that the physical modifications had been preliminarily reviewed and are not anticipated to change the LRA scope as discussed below. The licensee stated:

a) The majority of EPU modifications have not yet been implemented, and the final review of the modifications is not complete. A small number of Unit 1 EPU modification reviews had been completed at the time of the LRA (included in the annual updates) and were included in the LRA scope. Based on the final reviews to date and preliminary reviews of the EPU physical design changes, no impact to the aging management in the LRA is anticipated. The EPU modifications involve installation of new components and equipment which could have a positive impact on aging management. Components and equipment utilized for EPU modifications are of similar design as the original equipment and, to date, have not involved any new aging management issues nor are they likely to.

b) The review of the EPU conditions was performed as part of the LRA. The conclusions on operational characteristics reached in TVA letters dated February 19, 2004 (January 28, 2004 Meeting Follow-up), May 28, 2004 (as updated by letter dated August 5, 2004), and August 3, 2004, concerning effects of the EPU on LRA still remain valid for license renewal.

   As previously discussed in the August 3, 2004 submittal, the Browns Ferry LRA was developed in parallel with the EPU submittal. The February 19, 2004 letter provided additional information discussing the effects of EPU on the LRA. That letter concluded that all evaluations in support of the LRA were valid at the EPU power level, except those associated with Section 4.3, Metal Fatigue. LRA Sections 4.2 and 4.3 were later updated by the letters dated May 28, 2004 and August 5, 2004 to reflect EPU conditions. Attachment 1 of the August 3, 2004 letter identified the changing operational characteristics that potentially affected the identified aging mechanisms following EPU. These operational characteristics were reviewed for the systems affected, as delineated in Attachment 2 of that letter, to verify the license renewal application remains valid.

c) No additional components, materials, environments and aging effects/mechanisms have been identified as a result of the EPU application that are not included in the LRA.

As stated in item (a) above, the physical modifications associated with the EPU are not anticipated to impact the aging management based on preliminary reviews.

Final reviews will be completed post-implementation in accordance with TVA design processes and the aging management program. Also as stated in item (b) above, the review of EPU operating conditions was previously performed and appropriately incorporated into the LRA.

The NRC staff evaluated TVA's response and finds that a commitment to perform a final review of plant modifications to determine the impact on license renewal is appropriate to ensure that an aging management review will be performed. However, the staff was concerned that any new components or material/environment combinations to be installed to support the restart that were not included in the LRA should be identified now and evaluated for aging management as part of this review rather than waiting for the annual update. To resolve this concern, the licensee subsequently agreed to submit an update on the status of the modifications, and a final letter at the completion of all modifications confirming the information provided in the status letter.

On the basis of the additional information submitted by the licensee on December 1, 2006, and pending confirmation that the final review of modifications has been completed according to plant procedures, the NRC staff finds that there is reasonable assurance that plant changes have been adequately evaluated to confirm that appropriate aging management reviews have been performed.

Conclusion

The NRC staff has reviewed Appendix F to the LRA and the additional information provided by the licensee and concludes that the licensee (pending final review of EPU plant modifications) has adequately considered the impact of EPU modifications and changes in operating practices relative to license renewal. Therefore, there is reasonable assurance that SSC that have been modified to support the EPU will be subject to an aging management review.

3.0 RENEWED FACILITY OPERATING LICENSE AND TS CHANGES

To achieve an uprate of 5 percent, the licensee proposed the following changes to the Renewed Facility Operating License and TSs for Unit 1.

Changes to Browns Ferry Unit 1 Renewed Operating License and TS		
Section	Title	Description of Change
License Condition 2.C.(1)	Maximum Power Level	Revise the value of the Maximum Power Level to the uprated power level of 3458 MWt.
TS 2.1.1	Definitions - RATED THERMAL POWER	Revise the value of RTP definition to the uprated power level of 3458 MWt.
TS SR 3.1.7.5	SLC Surveillance	Revise minimum required amount of Boron-10 required from > 186 pounds to > 203 pounds.
TS SR 3.1.7.7	SLC Surveillance	Revise the SLC System pump discharge pressure requirement from 1275 to 1325 psig.
Table 3.3.1.1-1, Function 2.b	Flow Biased Simulated Thermal Power – High	Revise the Allowable Value formula for the Flow Biased Simulated Thermal Power – High.
Table 3.3.1.1-1, Note (b)	Flow Biased Simulated Thermal Power – High	Revise the Allowable Value formula Flow Biased Simulated Thermal Power - High for single loop
Table 3.3.1.1-1, Function 3	Reactor Vessel Steam Dome Pressure scram	Revise the Allowable Value for the Reactor Vessel Steam Dome Pressure scram function from ≤ 1055 psig to ≤ 1090 psig.
TS SR 3.3.4.2.3	ATWS-RPT Reactor Steam Dome Pressure	Revise the ATWS-RPT Reactor Steam Dome Pressure from 1146.5 psig to 1175 psig.
TS SR 3.4.3.1	SRV Surveillance	Revise the SRV setpoint pressures from 1105, 1115, and 1125 psig to 1135, 1145, and 1155 psig, respectively.
TS LCO 3.4.10	Reactor Steam Dome Pressure	Revise the Reactor Steam Dome pressure LCO from ≤ 1020 psig to ≤ 1050 psig.
TS SR 3.4.10.1	Reactor Steam Dome Pressure Surveillance	Revise the SR Reactor Steam Dome Pressure from ≤ 1020 psig to ≤ 1050 psig.
TS SR 3.5.1.7	ECCS Surveillance	Revise the HPCI surveillance test reactor pressure range from ≤ 1010 psig and ≥ 920 psig to ≤ 1040 psig and ≥ 950 psig.

Changes to Browns Ferry Unit 1 Renewed Operating License and TS		
TS SR 3.5.3.3	ECCS Surveillance	Revise the RCIC surveillance test reactor pressure range from ≤ 1010 psig and ≥ 920 psig to ≤ 1040 psig and ≥ 950 psig.
5.5.12	Primary Containment Leakage Rate Testing Program	Revise the peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, to 48.5 psig.

TS 2.1.1 Reactor Core Safety Limits (SLs)

With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow, the limit on the core thermal power will be changed from 25% to 23% of rated thermal power. The TS change reflects the proposed change in the plant and it is consistent with the results of the licensee's supporting safety analyses. The NRC staff finds this proposed change acceptable.

TS 3.1.7 Standby Liquid Control System

(a) SR 3.1.7.5, Minimum quantity of Boron-10 in the SLC solution tank and available for injection is changed from 186 lb to 203 lb. The minimum reactor boron concentration is increased from 660 ppm to 720 ppm due to the core design changes. The minimum quantity of boron specified in the TS SR 3.1.7.4 is increased from 186 lb to 203 lb due to increase in boron concentration. The SLC system shutdown capability is reevaluated for each reload core.

(b) SR 3.1.7.7, Pump Discharge Pressure of 1275 PSIG for the verification of the flow rate is increased to 1325 psig. The licensee performed a plant-specific ATWS analysis. The peak calculated vessel pressure during SLCS operation is 1204 psig for the limiting event. This equates to the pump discharge pressure of 1298 psig for the limiting ATWS case. The proposed pump discharge pressure of 1325 psig given in the TS SR 3.1.7.6 bounds this value.

TS Table 3.3.1.1-1, Function 2b and Note b., Average Flow range Monitors - Flow Biased Simulated Thermal Power - High

The flow-biased APRM simulated thermal power monitor analytical limit (AL) and scram setpoints are lowered proportionally to the increase in rated power, such that they remain substantially unchanged in terms of absolute power and core flow. The AV for the APRM flow-biased simulated thermal power - high scram has been changed from ≤ 0.58 W + 66% RTP to ≤ 0.66 W + 66%. The licensee has changed the footnote (b) to show the revised value for single loop operation. This proposed change ensures the pre-uprate design margins and licensing basis are preserved for operation at the uprated power. Since the licensee used the NRC staff-approved setpoint methodology for this function, the NRC staff finds the proposed AV acceptable.

The licensee has not taken any credit in the transient and accident analysis for this function in ensuring safety limits are met and therefore, this function is not considered to be a limiting safety system setting. Also, the as-left values will always be verified to be equal to the nominal trip setpoint. If an instrument function setpoint does not meet a TS AV, the Plant Procedure SPP-6.7, Instrument Setpoint, Scaling, and Calibration Program, gives direction for controlling out-of-calibration instrument conditions and contains the requirements for entering the issue into the corrective action program. A failure to meet the surveillance criteria is documented in the surveillance test data package and will be remedied prior to returning the instrument to operation.

Based on the review of the information provided by the licensee, the NRC staff finds that the setpoint calibration procedures maintain the trip setpoints within the established setting tolerance to ensure that the instruments will be capable of performing their specified safety functions and therefore, are acceptable.

TS Table 3.3.1.1-1, Function 3, Reactor Vessel Steam Dome Pressure - High

The RV steam dome pressure scram limit is increased because the steam dome operating pressure is increased. Operating pressure for power uprate is increased to assure that satisfactory reactor pressure control is maintained. The setting permits normal operation without spurious scrams, yet provides adequate margins to the maximum allowable RV pressure. The high-pressure scram terminates a pressurization transient not terminated by direct/primary scram (e.g., turbine stop valve closure) or high neutron flux scram. The licensee has proposed and the NRC staff has accepted that this function is safety-limit related in a license amendment (TS Change-453), which was issued on September 14, 2006. The licensee has added the acceptable footnote to the TS Table for this function. The proposed change to AV is acceptable to the NRC staff as licensee has used the NRC staff approved instrument setpoint methodology.

TS Section 3.3.4.2, Surveillance Requirement 3.3.4.2.3.b

The ATWS-RPT high pressure setpoint initiates trip of the recirculation pumps, thereby adding negative reactivity following events in which a scram does not occur even though it should have. The AL for ATWS-RPT high pressure setpoint was increased by 30 psi to account for the 30-psi increase in vessel operating pressure, SRV setpoints, etc. The analyses demonstrate that the ATWS criteria are met with the higher AL. Therefore, the higher AV is consistent with the ATWS safety analysis. This also prevents the unnecessary RPT following pressure transients with reactor scram, which helps in better mixing of the reactor coolant and reduces thermal stratification in the vessel. Since the licensee has used the NRC staff's approved setpoint methodology for this function, the changes to AV are acceptable. Also, because the ATWS-RPT is used for ATWS events and does not affect DBA analysis, it is not considered a safety limit related function. The setpoint calibration procedures maintain the trip setpoints within the established setting tolerance to ensure that the instruments will be capable of performing their specified safety functions and therefore, are acceptable.

TS 3.4.3 Safety Relief Valves

To verify that the safety function lift settings of the required 12 SRVs are within ± 3 percent of the setpoint as follows:

Number of SRVs	Setpoint (psig)
4	Increased from 1105 to 1135
4	Increased from 1115 to 1145
5	Increased from 1125 to 1155

Following testing, lift settings shall be within ± 1 percent. The TS change reflects the proposed change in the plant and it is consistent with the results of the licensee's supporting safety analyses. The NRC staff finds this proposed change acceptable.

TS 3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 and SR 3.4.10.1 will be changed to reflect the increase operating dome pressure from 1020 to 1050 psig. The TS change reflects the proposed change in the plant and it is consistent with the results of the NRC staff's review as described in section 2.8.4.2. The NRC staff finds this proposed change acceptable.

TS 3.5.1 ECCS Operating, SR 3.5.1.7, SR 3.5.3.3

The proposed reactor pressure for HPCI and RCIC pump tests will be changed to reflect the increase operating dome pressure from 1010 to 1040 psig. The pressure increases from 1010 to 1040 psig and 920 to 950 psig is acceptable as described in sections 2.8.4 and 2.8.5.

TS 5.5.12 Primary Containment Leakage Rate Testing Program

P_a is the pressure at which containment leakage rate testing is performed. It is defined in 10 CFR Part 50, Appendix J as the calculated peak containment internal pressure related to the design basis LOCA. The licensee proposes to revise P_a in TS 5.5.12 to 48.5 psig. The NRC staff finds this acceptable since P_a, the calculated peak containment internal pressure related to the design basis LOCA for the EPU, is determined with acceptable methods and assumptions and the value is bounding for power uprate conditions.

4.0 REGULATORY COMMITMENTS

In the September 22, 2006, request for interim approval of an increase in rated thermal power to 105 percent and other applicable documents, the licensee has made the following regulatory commitments:

- As discussed in Section 2.2.4, the licensee stated that the MOV switch settings at Unit 1 would be set prior to restart, but that some dynamic testing would be conducted during power ascension. TVA also indicated its commitment to implement the Joint Owners' Group Program on MOV periodic verification as part of its response to GL 96-05.

- As discussed in Section 2.5.3.1, the licensee has established a regulatory commitment to implement procedure changes that will: (1) define and control the generation of cycle-specific fuel pool heat load calculations, and (2) control the installation of the fuel pool gates based on the calculated fuel pool heat load.

- As discussed in Section 2.6.5, the licensee has committed to terminate drywell cooling within 2 hours of entry into the safe shutdown procedure which would be used for a shutdown due to fire.

- As discussed in Section 2.8, TVA committed to provide a new Supplemental Reload Licensing Topical Report for 105-percent specifically addressing the SLMCPR issue before January 31, 2007. The report was provided on January 29, 2007.

- As part of the two-step approach to EPU, TVA will perform two LTTs:

  - A large transient test that simulates the rejection of generator load will be completed within 30 days of reaching 105-percent OLTP.

  - An MSIV Closure with valve position scram large transient test will be performed within 30 days of reaching 115 percent to 120-percent OLTP.

  As discussed in Section 2.12, the NRC staff reviewed this commitment and did not agree with the power level proposed for the second LTT, therefore a License Condition was established mandating performance at 105 percent.

- As discussed in Section 2.12, TVA committed to trip a condensate booster pump, a condensate pump, and a main feedwater pump on an individual basis (i.e., one at a time). The NRC staff reviewed this commitment and issued a license condition mandating performance at 105 percent.

- As discussed in Section 2.1.3.1, the NRC staff was concerned that any new components or material/environment combinations to be installed to support the restart that were not included in the LRA should be identified and evaluated for aging management as part of this review rather than waiting for the annual update. To resolve this concern, the licensee agreed to submit an update on the status of the modifications. This letter was submitted on December 1, 2006.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are best provided by the licensee's administrative processes, including its commitment management program. Unless otherwise identified, the above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

5.0 RECOMMENDED AREAS FOR INSPECTION

As described above, the NRC staff has conducted an extensive review of the licensee's plans and analyses related to the proposed uprate and concluded that they are acceptable. The NRC staff’s review has identified the following areas for consideration by the NRC inspection staff during the licensee's implementation of the proposed uprate. These areas are recommended based on past experience with uprates, the extent and unique nature of modifications necessary to implement the proposed uprate, and new conditions of operation necessary for the proposed uprate. They do not constitute inspection requirements, but are intended to give inspectors insight into important bases for approving the uprate.

Section 2.2.4 - TVA described the modifications planned for Unit 1 in support of the EPU request. Many of the modifications are related to the changes in MS and FW operating parameters. For example, condensate, condensate booster, and reactor FW modifications are being performed to upgrade the components to provide the higher flows for EPU operating conditions. Many of the GL 89-10 MOVs will be replaced with the remainder being refurbished. The NRC staff will review the GL 89-10 MOV modifications prior to restart of Unit 1.

Section 2.3.2.2 - For uprate conditions, the generator hydrogen operating pressure for Unit 1 will be increased to the existing design pressure rating of 75 psig. The hydrogen pressure regulators and associated setpoints will be adjusted for 75 psig operation.

Section 2.5.1.4 - During an inspection in September 2006, the NRC reviewed the licensee’s fire protection procedures for Unit 1. The inspection staff found that an evaluation of the time needed to perform operator manual actions had been satisfactorily completed. However, as the safe shutdown procedures for Unit 1 are currently incomplete, in that the feasibility to perform operator manual actions has not yet been completed, the NRC inspection staff will confirm sufficient time is available for the operator to perform the necessary actions.

Section 2.11 - The licensee indicated that simulator and classroom training will be completed during the last training phase prior to restart. This training should include normal operating procedure actions required to achieve power uprate, power ascension testing, and physical plant changes as modeled in the simulator.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft Environmental Assessment and finding of no significant impact was prepared and published in the *Federal Register,* November 6, 2006 (71 FR 65009). The draft Environmental Assessment provided a 30-day opportunity for public comment. No comments were received on the draft Environmental Assessment. The final Environmental Assessment was published in the *Federal Register* on February 12, 2007 (72 FR 6612). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

8.0 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. This interim approval used the information supplied in support of the June 28, 2004, EPU application to determine if there is reasonable assurance that operation at uprate conditions is consistent with the Commission's regulations.

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Attachment: Acronym List

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LIST OF ACRONYMS

ac	alternating current
ADAMS	Agencywide Documents Access and Management System
ADHRS	Auxiliary Decay Heat Removal System
ADS	automatic depressurization system
AEC	Atomic Energy Commission
AL	analytical limit
ALARA	as low as reasonably achievable
AMP	aging management program
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOI	Abnormal Operating Instructions
AOO	anticipated operational occurrence
AOV	air-operated valve
APRM	average power range monitor
ARI	alternate rod injection
ASME	American Society of Mechanical Engineers
AST	alternate source term
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AV	allowable value
BFN	Browns Ferry Nuclear Plant
BL	Bulletin
BOP	balance-of-plant
BTP	branch technical position
BWR	boiling-water reactor
BWROG	Boiling Water Reactor Owners Group
BWRVIP	Boiling Water Reactor Vessel and Internals Project
C-15	Carbon-15
CAD	containment atmospheric dilution

CAP	containment-accident pressure
CD	confirmation density
CDF	core damage frequency
cfm	cubic feet per minute
CFR	*Code of Federal Regulations*
CFS	condensate and feedwater system
CLTP	current licensed thermal power
COLR	core operating limits report
CRAVS	control room area ventilation system
CRD	control rod drive
CRDM	control rod drive mechanism
CRDS	control rod drive system
CS	core spray
CST	condensate storage tank
CUF	cumulative usage factor
CWS	circulating water system
DBA	design-basis accident
DBLOCA	design-basis loss-of-coolant accident
dc	direct current
DG	diesel generator
DGB	diesel generator building
DSS/CD	Detect and Suppress Solution/Confirmation Density
ECCS	emergency core cooling system
EDG	emergency diesel generator
EECW	Emergency Equipment Cooling Water
EFDS	equipment and floor drainage system
EFPY	effective full-power year
EHC	electro-hydraulic control
ELTR	General Electric Topical Report
EOI	Emergency Operating Instruction
EOP	Emergency Operating Procedures

EOS	emergency overspeed
EPG	emergency procedure guideline
EPGs/SAGs	emergency procedure guidelines/severe accident guidelines
EPRI	Electric Power Research Institute
EPU	extended power uprate
EQ	environmental qualification
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
ESFVS	engineered safety feature ventilation system
FAC	flow-accelerated corrosion
FFWTR	final feedwater temperature reduction
FHA	fuel-handling accident
FIV	flow-induced vibration
FPP	fire protection program
fps	feet per second
FR	*Federal Register*
FUSAR	Framatome Uprate Safety Analysis Report
FW	feedwater
FWCF	feedwater controller failure
FWH	feedwater heater
FWHOOS	feedwater heater out of service
GDC	general design criterion / criteria
GE	General Electric Company
GL	Generic Letter
gpm	gallons per minute
GWMS	gaseous waste management system
HWWV	hardened wetwell vent
HPCI	high-pressure coolant injection
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and controls
IASCC	irradiation assisted stress corrosion cracking

ICA	interim corrective action
IGSCC	intergranular stress corrosion cracking
IPE	individual plant examination
IPEEE	individual plant examination of external events
ISP	integrated surveillance program
ISI	inservice inspection
IST	inservice testing
ksi	kilo-pounds per square inch
kV	kiloVolt
lb	pound
LCO	limiting condition for operation
LERF	large early release frequency
LFWH	loss of feedwater heating
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LOFW	loss of feedwater
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LPZ	low population zone
LRA	license renewal application
LRNBP	load rejection, no bypass
LTR	Licensing Topical Report
LTSS	long-term stability solution
LTT	large transient test
LWMS	liquid waste management system
MAPLHGR	maximum average planar linear heat generation rate
MBT	Main Bank Transformer
MCES	main condenser evacuation system
MCPR	minimum critical power ratio
MOV	motor-operated valve
MS	main steam

MSIV	main steam isolation valve
MSIVLCS	main steam isolation valve leakage control system
MSL	main steam line
MSLB	main steam line break
MSRV	main steam relief valve
MSSS	main steam supply system
MTSV	master trip solenoid valve
MVA	mega volt amps
MWt	megawatts thermal
N-16	nitrogen-16
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
NUMAC	Nuclear Measurement, Analysis, and Control
O&M	operations and maintenance
OLMCPR	operating limit minimum critical power ratio
OLTP	original licensed thermal power
OOS	out of service
OPRM	oscillation power range monitor
PBDA	Period Based Detection Algorithm
PCT	peak cladding temperature
ppm	parts per million
PRNM	power range neutron monitoring
psi	pounds per square inch
psia	pounds per square inch absolute
psid	pounds per square inch differential
psig	pounds per square inch gauge
PSO	Power Systems Operations
P-T	pressure-temperature

PUSAR	Power Uprate Safety Analysis Report
RAI	request for additional information
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RHRSW	residual heat removal service water
RMI	reflective metallic insulation
RPS	reactor protection system
RPT	recirculation pump trip
RPV	reactor pressure vessel
RRS	reactor recirculation system
RS	Review Standard
RSTP	restart test program
RS-001	*Review Standard for Extended Power Uprates*
RTP	rated thermal power
RV	reactor vessel
RWCS	reactor water cleanup system
RWCU	reactor water cleanup
SAFDL	specified acceptable fuel design limit
SAG	severe accident guideline
SAR	Safety Analysis Report
SBO	station blackout
SCC	stress corrosion cracking
SDC	shutdown cooling
SE	safety evaluation
SFP	spent fuel pool
SFPAVS	spent fuel pool area ventilation system
SFPCS	spent fuel pool cooling system
SGTS	standby gas treatment system

SLC	standby liquid control
SLCS	standby liquid control system
SLMCPR	safety limit minimum critical power ratio
SOV	solenoid-operated valve
SPC	suppression pool cooling
SPDS	safety parameter display system
SR	Surveillance Requirement
SRLR	Supplemental Reload Licensing Report
SRP	Standard Review Plan
SRV	safety relief valve
SSCs	structures, systems, and components
SWS	service water system
TAF	top of active fuel
TDH	total dynamic head
TEDE	total effective dose equivalent
TGSS	turbine gland sealing system
TR	topical report
TRM	technical requirements manual
TS	technical specification
TSBS	turbine steam bypass system
TTNBP	turbine trip, no bypass
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
UHS	ultimate heat sink
USAS	United States of America Standards
USE	upper shelf energy

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