February 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 CYCLE-SPECIFIC SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (TAC NO. MD1721) (TS-455)

Dear Mr. Singer:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 267 to Renewed Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant (BFN), Unit 1. This amendment is in response to your application dated May 1, 2006, as supplemented by letters dated September 1 and November 6, 2006. This amendment revises the numeric values of the safety limit minimum critical power ratio (SLMCPR) in the Technical Specification Section 2.1.1.2 for one and two reactor recirculation loop operation to incorporate the results of the BFN, Unit 1, Cycle 7, SLMCPR analysis.

BFN Unit 1, Cycle 7, SLMCPR analysis was performed based on extended power uprate conditions. Approval of this amendment does not constitute authority to operate above the current licensed thermal power.

The non-proprietary version of the NRC's safety evaluation related to this amendment is enclosed. The proprietary version is being transmitted under separate cover. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Margaret H. Chernoff, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-259

Enclosures: 1. Amendment No. 267 to
License No. DPR-33
2. Safety Evaluation (nonproprietary)

cc w/enclosures: See next page

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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. 267
Renewed License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 1, 2006, as supplemented by letters dated September 1, 2006, and November 6, 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 Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 267, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Patrick D. Milano for/

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: February 6, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 267

RENEWED FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Replace Page 3 of Renewed Operating License DPR-33 with the attached Page 3.

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
2.0-1	2.0-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 267

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

1.0 INTRODUCTION

By application dated May 1, 2006, as supplemented by letters dated September 1, 2006 and November 6, 2006, the Tennessee Valley Authority (TVA, the licensee) submitted a request for changes to the Browns Ferry Nuclear Plant (BFN), Unit 1, Technical Specifications (TSs). The supplements dated September 1, and November 6, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 15, 2006 (71 FR 46937). The requested changes revise the numeric values of the safety limit (SL) minimum critical power ratio (MCPR) (SLMCPR) in the TS Section 2.1.1.2 for one and two reactor recirculation loop operation to incorporate the results of the BFN, Unit 1 Cycle 7, SLMCPR analysis.

By letter dated June 28, 2004, TVA submitted a request to increase BFN, Unit 1, thermal power to approximately 120 percent of the current thermal power rating. BFN, Unit 1, Cycle 7, SLMCPR analysis was performed based on extended power uprate (EPU) conditions. Approval of this amendment does not constitute authority to operate above the current licensed thermal power.

On September 22, 2006, TVA submitted a request to supplement its BFN, Unit 1, June 28, 2004, letter. This supplement requested interim approval of an increase in thermal power of approximately 5-percent original licensed thermal power. The licensee stated in its September 22, 2006, submittal, that all safety analyses for operation at 105-percent original licensed thermal power were acceptably bound by previous analyses performed assuming Cycle 7 operation at 120-percent original licensed thermal power. While the NRC staff concluded, in most cases, that this assumption was acceptable, the NRC staff identified several concerns with regard to prolonged changes in operating strategies that could affect the SLMCPR in a nonconservative manner. Therefore, the NRC staff requested the licensee to supplement its SLMCPR calculation. The supplement, submitted by letter dated November 6, 2006, specifically addressed the concerns identified by the NRC staff.

2.0 EVALUATION

2.1 Description of Change

The licensee for BFN Unit 1 proposes to revise TS 2.1.1.2 for the dual recirculation loop and single recirculation loop SLMCPR values to reflect results of a cycle-specific calculation performed by Global Nuclear Fuels (GNF) for BFN Unit 1 Cycle 7 operation. Specifically, the licensee seeks to change the SLMCPR from 1.10 to 1.09 for dual-loop recirculation, and from 1.12 to 1.11 for single-loop recirculation, at steam dome pressures greater than 785 pounds per square inch gauge (psig) and at core flows greater than 10 percent of rated core flow.

The licensee shut down BFN Unit 1 voluntarily in 1985, and it has not operated since. As a result, the Cycle 7 core is an initial core designed for the restart of BFN Unit 1. Cycle 7, BFN Unit 1 will also implement operation at a higher power/flow operating domain known as the Maximum Extended Load Line Limit Analysis (MELLLA). The Cycle 7 core is designed for a nominal 24-month EPU operating cycle.

The BFN Unit 1, Cycle 7, initial core consists of predominantly fresh fuel. The EPU/MELLLA core is loaded with 564 fresh GE [General Electric]14 fuel assemblies, 108 fresh GE13 fuel assemblies, and 92 previously-irradiated assemblies from the BFN Unit 2, Cycle 13 core. The irradiated fuel is loaded in the periphery of the core, and the fresh fuel is loaded in the core interior. This fuel-loading pattern makes the BFN Unit 1 core design unique for upcoming Cycle 7, in which the core will be loaded with high reactivity once-burned fuel.

The SLMCPR is calculated on a cycle-specific basis, because it is necessary to account for the core configuration-specific neutronic and thermal-hydraulic response.

2.2 Regulatory Evaluation

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, “Technical specifications," provides the regulatory requirements for the content required in a licensee’s TSs. As stated in 10 CFR 50.36, the TSs will include Surveillance Requirements (SRs) to assure that the limiting conditions for operation (LCO) will be met.

At the time BFN was licensed, 10 CFR Part 50 Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," was not incorporated in the NRC regulations. BFN conformed to the draft Proposed GDC 27 (Units 1 and 2) and draft Proposed GDC 70 (Unit 3) criteria current at the time of the BFN design. The design bases of each unit of this plant were reevaluated by the licensee against the draft Proposed GDC 70 criteria current at the time of operating license application. The licensee concluded that each unit of this plant conforms with the intent of the GDC for Nuclear Power Plant Construction Permits. The references to the criteria below refer to these proposed GDC, not the GDC in Appendix A of 10 CFR Part 50.

Draft proposed Criterion 6, “Reactor core design,” states that the reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits that have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations that can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

NUREG-0800, "Standard Review Plan for the Review of Safety Analyses Reports for Nuclear Power Plants," provides guidance to NRC staff reviewers in the Office of Nuclear Reactor Regulation in performing safety reviews of applications to construct or operate nuclear power plants and the review of applications to approve standard designs and sites for nuclear power plants. NUREG-0800, Section 4.4, "Thermal and Hydraulic Design," states that the critical power ratio (CPR) is to be established such that at least 99.9 percent of fuel rods in the core would not be expected to experience departure from nucleate boiling or boiling transition during normal operation or anticipated operational occurrences.

3.0 TECHNICAL EVALUATION

Fuel design limits can be exceeded if the core exceeds critical power. Critical power is a term used for the power at which the fuel departs from nucleate boiling and enters a transition to film boiling. For boiling-water reactors, the critical power is predicted using a correlation known as the GE critical quality boiling length correlation, commonly known as the GEXL correlation. Due to core-wide and operational variations, the margin to transition boiling is most easily described in terms of a CPR, which is defined as the rod critical power as calculated by GEXL, divided by the actual rod power. The SLMCPR is calculated using a statistical process that takes into account all operating parameters and associated uncertainties.

The MCPR fuel cladding integrity SL ensures that during normal operation and during anticipated operational occurrences, at least 99.9 percent of the fuel rods in the core do not experience transition boiling. This is accomplished by the determination of a CPR margin for transients, which is added to the SLMCPR to determine the operating limit MCPR (OLMCPR). At the OLMCPR, at least 99.9 percent of the fuel rods would be expected not to experience boiling transition during normal operations and transients caused by single operator error or equipment malfunction.

The licensee described the methodologies used to calculate the SLMCPR value for the proposed TS change in the submittal. The Cycle 7 SLMCPR analysis was performed by GNF using plant- and cycle-specific fuel and core parameters.

GNF performed the analysis using NRC-approved methodologies NEDC-32601P-A, “Methodology and Uncertainties for Safety Limit MCPR Evaluations,” NEDC-32694P-A, “Power Distribution Uncertainties for Safety Limit MCPR Evaluations,” and Amendment 25 to NEDE-24011, “General Electric Standard Application for Reactor Fuel."

By letter dated March 11, 1999, the NRC approved these methodologies subject to the following applicable conditions:

(1) Fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons, since changes in fuel design can have a significant effect on calculation accuracy.

(2) The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of the R-Factor uncertainty when the methodology is applied to a new fuel lattice.

(3) The 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons in Tables 3.1 and 3.2 of NEDC-32601P-A.

These conditions were addressed in letters to the NRC from Glen Watford of GNF: "Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR, Power Distribution and R-Factor Methodologies," dated September 24, 2001, and "Confirmation of the Applicability of the GEXL14 Correlation and Associated R-Factor Methodology for Calculating SLMCPR Values in Cores Containing GE14 Fuel," dated October 1, 2001. The NRC staff has concluded that these letters indicate that GNF has satisfied the three restrictions and that, in this respect, the methodologies remain acceptable for the BFN Unit 1 Cycle 7 core.

By letter dated November 6, 2006, TVA addressed several concerns identified by the NRC staff 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 fuel rods near boiling transition in the limiting SLMCPR scenario.

The SLMCPR re-analysis performed by GNF, the licensee's fuel vendor, specifically considered axial power shapes, and concluded that potential limiting axial power shapes were not present, therefore, no power shape penalties were applied to the calculated BFN, Unit 1, Cycle 7 SLMCPR values.

The NRC staff finds this determination acceptable because its supporting analysis was performed in accordance with the NRC-approved methodologies. Therefore, no additional conservatism is necessary beyond the proposed 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. GNF 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 can provide an acceptable SLMCPR value. The NRC staff also confirmed that the SLMCPR analyzed by GNF for this purpose was less than the proposed SLMCPR, therefore, the proposed SLMCPR is more conservative. Therefore, the NRC staff finds that no additional conservatism for the proposed SLMCPR values is necessary for changes in power distribution due to operation below 120 percent of original licensed thermal power.

3.1 Confirmation of Power Distribution Uncertainties for EPU Conditions

The review of the BFN Unit 1 EPU includes confirmation that the power distribution uncertainties applied to the thermal limits analyses remain valid and applicable for the EPU neutronic and thermal-hydraulic conditions. The power distribution uncertainties applied to the SLMCPR calculations are specified in NEDC-32601P-A and NEDC-32694P-A. Restrictions applied to the methodology described in the topical reports require confirmation of the uncertainties with new fuel design and operating strategy changes. Review of the SLMCPR methodology indicates that the 4 bundle power allocation P_{pal} and the peak pin uncertainty P_{peak} have not been confirmed through pin and bundle power gamma scans for the GE14 fuel as currently operated.

Since the initial qualification of the steady-state neutronic methods in NEDE-30130P-A, "Steady State Nuclear Methods," dated May 1985, GE Nuclear Energy (GENE) has not performed any gamma scans to benchmark the codes' adequacy in predicting the bundle and pin powers for the current fuel designs and for the current operating strategies. Without measurement data, the neutronic methods' capability to predict bundle and pin powers, or the impact of depletion at high void conditions, can neither be assessed, nor the uncertainties established.

The fuel vendor relies heavily on traversing incore probe (TIP) measured/calculated four bundle power peaking and code-to-code comparisons. Recently, the fuel vendor had compiled comprehensive core follow TIP comparisons from plants operating with uprated power levels and/or high power densities. However, core follow data, while useful for monitoring core performance, does not form adequate bases for qualifying neutronic code systems after implementing substantial changes in the core operating strategies and fuel designs. Section 5.2 of NEDE-30130P-A compared the relative merits of using TIP comparisons (core follow) for validating neutronic code systems, stating, "the TIP signals provide a good picture of the axial power distribution, but do not provide a detailed bundle by bundle distribution, because there is only one TIP location for every 16 bundles. A more accurate estimate of the reactor power distribution can be obtained just prior to a reactor shutdown by the procedure known as gamma scanning . . ."

In light of the above statement, TVA referenced NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," in their EPU application for Cycle 7. The topical report provides assessment of the applicability of GENE, NRC-approved, analytical methods and codes used to support operation at EPU conditions at BFN Unit 1. The topical report covered the power distribution uncertainties applied to the SLMCPR calculation and the supporting measurement data. The report proposed applying a statistical treatment to currently available gamma scan data, which has resulted in an increase in power distribution uncertainties applied to the SLMCPR. The pin power uncertainties change from [ ] percent to [ ] percent. The bundle power allocation uncertainty (σ_{pal}) changes from [ ] percent to [ ] percent. The increase in the pin and bundle power uncertainties yield a total of 0.02 increase of the SLMCPR.

Accordingly, for BFN Unit 1, the cycle-specific SLMCPR value has been increased by a value of 0.02. This SLMCPR adder will provide additional confidence that the power distribution uncertainties applied to the SLMCPR remains conservative.

3.2 Power Shape Penalties

Based on the findings of an NRC staff audit of the GNF GEXL databases, the NRC staff requires plant specific justification for the validity of the GEXL correlation. In the case of BFN, Unit 1, [[

]] The NRC staff finds that penalties were not necessary at BFN, Unit 1, and that the GEXL correlation remains valid based on the above consideration.

3.3 Deviation from R-Factor Uncertainties

The licensee proposes to deviate from R-factor uncertainties that have previously been accepted by the NRC. The R-factor is an input to the GEXL critical power correlation that captures the local peaking influence on the predicted onset of boiling transition. The R-factor uncertainty is related to the uncertainty associated with nuclear methods in determining the fuel pin power peaking, and it includes terms for manufacturing and channel bow uncertainties. [[

]] The NRC has reviewed this approach, and found it to influence the SLMCPR in a conservative manner. The NRC staff also considered that BFN, Unit 1 is a D lattice plant and, therefore, is not susceptible to control blade shadow corrosion induced channel bow. Finally, the NRC staff confirmed that the fuel channels used in the BFN Unit 1, Cycle 7, core are either new or once-burned. Therefore, the NRC staff finds that this deviation is both conservative and acceptable.

3.4 Zircaloy Spacer Issues

The GEXL14 correlation used to determine the SLMCPR for the GE14 fuel was developed using test results at the ATLAS test facility. Communications submitted by GE in accordance with 10 CFR Part 21 indicated a potential problem with the GEXL14 correlation due to GE14

Zircaloy spacer spring deformation during critical power testing at the ATLAS facility. As a result of this deformation, GE determined that the GEXL14 correlation may cause licensees to determine CPRs in a nonconservative manner. As a corrective action, licensees were requested to apply conservative additive constants to susceptible rod locations, which modified the R-factors used in the GEXL14 correlation. The NRC staff reviewed the licensee's submittal in light of the spring deformation issue and determined that the licensee has applied the conservatisms in its critical power determinations in a manner consistent with GE recommendations for its operating licensees. The NRC staff finds this approach acceptable.

3.5 Prior Cycle Comparison

Since BFN, Unit 1 has not operated since 1985, and the core is comprised of mostly new fuel, the NRC staff has determined that the typical comparison to previous cycles that is customary of a SLMCPR evaluation is not warranted for BFN, Unit 1, Cycle 7.

The licensee will not be re-using stored control blades operated in Cycle 6. In response to the EPU Round 8 Request for Additional Information, dated August 18, 2006, TVA confirmed that BFN, Unit 1 will be operating with new control blades, which ensures that reactivity control will not be compromised due to control rod blade leaching or degradation.

In consideration of the above information, the NRC staff has determined that the licensee's proposed amendment to update the TSs to include cycle-specific SLMCPR is acceptable. The licensee is authorized to change the SLMCPR as existing in TS 2.1.1.2 from 1.10 to 1.09 for dual-loop recirculation, and from 1.12 to 1.11 for single-loop recirculation, at steam dome pressures greater than 785 psig and at core flows greater than 10 percent of rated core flow.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (71 FR 46937). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Benjamin T. Parks

Date: February 6, 2007

Mr. Karl W. Singer
Tennessee Valley Authority

BROWNS FERRY NUCLEAR PLANT

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