September 27, 2006

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: BROWN FERRY NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING POWER RANGE NEUTRON MONITOR UPGRADE (TAC NO. MC1330) (TS-430)

Dear Mr. Singer:

The Commission has issued the enclosed Amendment No. 262 to Renewed Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1, in response to your application dated November 10, 2003, as supplemented by letter dated November 8, 2004. This amendment incorporates the necessary Technical Specification (TS) changes for the planned replacement of the power range monitoring portion of the existing Neutron Monitoring System with a digital upgrade. These changes expand the current allowable operating domain to the Maximum Extended Load Line Limit region of the core power/flow chart.

The analyses used by the NRC staff in its review of the Maximum Extended Load Line Limit Analysis (MELLLA) were performed at a power level of 3952 MWt (120 percent of current licensed thermal power (CLTP)). The Nuclear Regulatory Commission (NRC) staff finds that the MELLLA conditions would remain bounding up to CLTP. Approval of this amendment does not constitute authority to operate above the CLTP.

By letter dated January 6, 2006, TVA submitted a proposed TS change involving the activation of thermal-hydraulic stability monitoring instrumentation. The proposed changes related to the activation of this instrumentation contained in the above submittals will be addressed by the NRC staff in our review of the January 6, 2006, submittal, and are not approved in this amendment.

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/

Margaret H. Chernoff, 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. 262 to
License No. DPR-33
2. Safety Evaluation

cc w/encls: See next page

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

Sincerely,

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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. 262
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 November 10, 2003, as supplemented November 8, 2004, 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. 262, 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 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Technical
Specifications

Date of Issuance: September 27, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 262

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
i	i
1.1-5	1.1-5
3.2-7	---
3.2-8	---
3.3-1	3.3-1
3.3-2	3.3-2
3.3-3	3.3-3
3.3-4	3.3-4
3.3-5	3.3-5
3.3-6	3.3-6
3.3-7	3.3-7
3.3-18	3.3-18
3.3-19	3.3-19
3.3-20	3.3-20
3.10-22	3.10-22
3.10-24	3.10-24

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 262 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 letter dated November 10, 2003 (Agencywide Documents Access and Management System (ADAMS) Number ML03300129), as supplemented by letter dated November 8, 2004 (ADAMS ML043130366), 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 application addresses design changes that would upgrade the analog power monitoring system in BFN Unit 1 with a General Electric Company (GE) Nuclear Measurement Analysis and Control Power Range Neutron Monitor System (NUMAC-PRNMS), including an Oscillation Power Range Monitoring (OPRM) function. It also proposes the implementation of the Average Power Range Monitor, Rod Block Monitor (RBM) TS improvements and operation in an expanded core power/flow domain, the Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) region. The supplement dated November 8, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 3, 2004 (69 FR 5208).

By letter dated January 6, 2006, TVA submitted a proposed TS change involving the activation of thermal-hydraulic stability monitoring instrumentation. The proposed changes related to the activation of the instrumentation contained in this application will be addressed by the Nuclear Regulatory Commission (NRC) staff in its review of the January 6, 2006, submittal, and are not approved in this amendment.

2.0 EVALUATION

2.1 Description of Change

BFN Unit 1 is a boiling-water reactor (BWR) 4 type reactor with a 251-inch diameter vessel and Mark I containment. The current licensed thermal power (CLTP) is 3293 megawatts-thermal. BFN Unit 1 is currently licensed and analyzed for increased core flow operation at 102.5 million pounds-mass per hour. The proposed change expands the current allowable operating domain to the Maximum Extended Load Line Limit region of the core power/flow chart. The currently analyzed limiting power/flow point for Single Loop Operation (SLO) will not change for the proposed MELLLA implementation since SLO is not extended into the MELLLA operating region.

The operational flexibility of a BWR during power ascension from the low-power, low-flow core condition to the rated high-power, high-flow core condition is restricted by several factors. Also, once rated power is achieved, periodic adjustments to core flow and control rod positions must be made to compensate for the reactivity changes due to Xenon buildup and decay, with fuel and burnable poison burnup. Factors currently restricting plant flexibility at BFN Unit 1 in efficiently achieving and maintaining rated power include:

1) the currently licensed allowable power/flow operating map; and,

2) the current Average Power Range Monitor (APRM) flow-biased flux scram and flow-biased rod block setdown requirements.

The licensee's proposed TS amendments applicable to the APRM portion of the BFN Unit 1 TSs are to be implemented following the replacement of the existing power range portion of the existing Neutron Monitoring System with a GE NUMAC-PRNMS, including the Option III OPRM function. The planned modification involves replacing the existing six APRM channels of power range monitor electronics with four channels of NUMAC-PRNMS hardware. The existing equipment is located in a five-bay panel in the main control room. The modification will remove and replace the existing power range monitor equipment within the panels but, with minor exceptions, leave the plant cabling and interfaces undisturbed.

All power range monitoring functions will be maintained in the new system, including the Local Power Range Monitor (LPRM) detector signal processing, LPRM averaging, APRM reactor trips, and RBM logic and interlocks. The LPRM input signals to the APRMs will be reconfigured such that each of the four new APRM channels will be allocated one-fourth of the existing LPRM signals.

The NUMAC-PRNMS consists of four APRM channels and four voter channels. Trip signals from each of the four APRM channels are sent to all four voter channels. One voter module is dedicated to each reactor protection system (RPS) trip relay. A reactor trip occurs when two or more of the four APRM functions or two or more of the four OPRM functions calculate a trip condition. The voters perform a vote of the OPRM channel trip outputs separate from the APRM trip outputs (i.e., an OPRM trip in one channel and an APRM trip in another channel will not result in a reactor trip from two of four voters in a trip state).

As part of the planned modification, the number of APRM instrument channels will be reduced from six channels to four channels. The LPRM inputs to the APRMs will also be reconfigured. The four APRM instrument channels will be combined in four 2-out-of-4 trip logic channels, which will provide input to the RPS trip channels.

The number of recirculation flow instrument channels associated with the APRMs will be increased from two total flow channels (four transmitters) to four total flow channels (eight transmitters). The recirculation flow signal processing, which was previously accomplished using separate hardware within the power range monitor control panels, will be integrated into the APRM chassis in the new PRNMS.

2.2 Regulatory Evaluation

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, “Technical specifications," which 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. The proposed TS changes would revise SRs and the LCO actions and completion times for each applicable operating condition.

10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," contains the regulatory requirements governing the ATWS. This regulation includes requirements for an ATWS Recirculation Pump Trip (RPT), an Alternate Rod Insertion system, and an adequate Standby Liquid Control System injection rate.

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems (ECCS) for light-water nuclear power reactors," provides acceptance criteria for ECCS cooling performance following postulated loss-of-coolant accidents (LOCAs).

At the time BFN was licensed, 10 CFR 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 70 (unit 3) criteria current at the time of the BFN design. The design bases of each unit of this plant were reevaluated against the draft 70 criteria current at the time of operating license application. It was 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 50.

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 which 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 which 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.

Criterion 7, “Suppression of power oscillations,” states that the core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

Criterion 40, "Missile Protection," states that protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Criterion 42, "Engineered safety features performance capability," states that engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

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.

Under certain conditions, BWRs may be susceptible to coupled neutronic/thermal-hydraulic instabilities. These instabilities are characterized by periodic power and flow oscillations. If these power and flow oscillations become large enough, the fuel cladding integrity minimum critical power ratio (MCPR) safety limit requirements may be challenged.

To detect core instabilities automatically and provide a reactor scram signal to the RPS, the licensee selected Boiling-Water Reactor Owners Group (BWROG) Stability Option III as the long-term stability system solution (LTSSS) for BFN Unit 1. The LTSSS Option III approach consists of detecting and suppressing stability-related power oscillations by automatically inserting control rods (scramming) to terminate power oscillations, consistent with proposed GDC criterion 6 and criterion 7.

The GE NUMAC-PRNMS design was approved by the NRC staff in its safety evaluation (SE) of GE Licensing Topical Report (LTR) NEDC-32410P-A, “Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function,” dated September 5, 1995. The SE was subsequently included in the approved version of the LTR (ADAMS ML9605290009). Supplement 1 of the LTR was approved by the NRC staff in its SE dated August 15, 1997. The SE was subsequently included in the approved Supplement 1 of the LTR (ADAMS ML9806120242). The LTR and Supplement 1 address the BWR power instability issue discussed in GE LTR NEDO-31960, “BWR Owners’ Group Long-Term Stability Solutions Licensing Methodology” (ADAMS ML9106100443).

The NRC staff approved use of the NUMAC-PRNMS with the OPRM functions in BWR design plants. The NUMAC-PRNMS with the Option III LTSSS function, when installed and operated in accordance with the approved guidance provided in the above-referenced licensing topical reports, is consistent with proposed GDC criterion 6 and criterion 7.

3.0 TECHNICAL EVALUATION

The licensee’s submittal contained two groups of requested TS changes. The changes identified as Group 1 are related to the proposed APRM and RBM TS (ARTS) improvements as a result of the NUMAC-PRNMS modification. The changes identified as Group 2 are related to concurrent implementation of improvements related to operations in the core power/flow domain, the MELLLA region. Information provided by TVA as part of the Extended Power Uprate (EPU) application dated June 28, 2004, related to the MELLLA was used in support of this review. The NRC staff finds this approach acceptable because MELLLA conditions at EPU bound the MELLLA conditions at CLTP. Approval of this amendment does not constitute authority to operate above the CLTP.

3.1.1 NUMAC-PRNMS Modification

The APRM system averages LPRM signals, processes flow signals from the reactor core recirculation flow instrumentation, and then compares the results to RPS trip set points. The OPRM detects and suppresses reactor core power instabilities using the Option III approach described in LTR NEDO-31960.

As stated in the NRC staff's SE of NEDC-32410P-A, to receive NRC approval of a NUMAC-PRNMS installation, the licensee must confirm the following:

1. The applicability of NEDC-32410P, including clarifications and reconciled differences between the specific plant design and the topical report design descriptions.

2. The applicability of the BWROG topical reports that address the NUMAC-PRNMS and associated instability functions, setpoints and margins.

3. Plant-specific revised TSs for the NUMAC-PRNMS functions are consistent with NEDC-32410P, Appendix H, and Supplement 1.

4. Plant-specific environmental conditions are enveloped by the NUMAC-PRNMS equipment environmental qualification values.

5. Administrative controls are provided for manually bypassing APRM/OPRM channels or protective functions, and for controlling access to the APRM/OPRM panel and channel bypass switch.

6. Any changes to the plant operator's panel have received human factors reviews per plant-specific procedures.

The licensee's actions with regard to the above conditions are discussed in the following sections.

3.1.1.1 Applicability of the NUMAC-PRNMS Design to the BFN Unit 1 Plant Design

The NRC staff compared the applicable BFN Unit 1 design features with the corresponding design features in LTR NEDC-32410P-A. BFN Unit 1 is a GE BWR/4, a BWR design addressed in the LTR. As stated in Section 2 above, the six APRM channels currently used in BFN Unit 1 will be combined into four 2-out-of-4 logic channels that will provide inputs through dedicated RPS channel voters to the four RPS channels. The licensee is increasing the number of recirculation flow instrument channels from two total flow channels (four transmitters) to four channels (eight transmitters). These proposed design modifications conform to the NUMAC-PRNMS design description in NEDC-32410P-A, and are compatible with the existing plant neutron monitoring system and RPS. Therefore, the NRC staff finds that the NUMAC-PRNMS design is applicable to BFN Unit 1.

3.1.1.2 PRNMS OPRM Instability Function Set Points, and Margins

The licensee will test the PRNMS instability function (OPRM), including the adequacy of the setpoint values and margins during the first fuel cycle of OPRM operation, using the methodology described in LTR NEDO-32465-A, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," dated August 1996 (ADAMS ML9609230137). The NRC staff approved the initial period for OPRM confirmatory testing in the SE of NEDC-32410P-A and, therefore, finds the licensee's approach for developing OPRM set points and margins acceptable.

3.1.1.3 Plant-specific TSs

See Section 3.1.3 for a discussion of the plant-specific revised TSs.

3.1.1.4 Plant-Specific Environmental Conditions

In Table 1, the BFN plant-specific environmental conditions for temperature, humidity, pressure, and radiation are compared to the NUMAC-PRNMS environmental qualification values.

Table 1. Comparison of BFN Unit 1 Environmental Conditions with NUMAC-PRNM Environment Qualification Values

	BFN Unit 1	NUMAC-PRNMS
Temperature	15.6°C to 40°C (60 °F to 104°F)	5°C to 50°C (41°F to 122°F)
Humidity	10% to 90% RH [relative humidity]	10% to 90% RH
Pressure	14.7 psia [pounds per square inch-absolute] to 14.72 psia	13 psia to 16 psia
Radiation	1E-3 Rads/hr dose rate 350 Rads TID [total integrated dose]	1E-3 Rads/hr dose rate 1E+3 Rads TID

As shown in Table 1, the BFN environmental conditions are enveloped by the NUMAC-PRNMS qualification values, and, therefore, are acceptable.

The NRC staff reviewed the seismic response spectra for the BFN units and concludes that the BFN Unit 1 seismic qualification is within the NUMAC-PRNMS seismic qualification envelope, and, therefore, is acceptable.

GE Licensing Topical Report NEDC-32410P-A states that new equipment and plant modifications should not produce unacceptable levels of noise emissions that could adversely affect NUMAC equipment, or the licensee is to take action to prevent these emissions from reaching potentially sensitive equipment. These measures apply for both noise susceptibility and emissions. The BFN design procedures require that all digital equipment systems installed or used within the plant be evaluated for susceptibility and emissions of electromagnetic interference (EMI) in accordance with the NRC approved Electric Power Research Institute Guideline TR-102323. The NRC staff finds this approach acceptable for ensuring the EMI environment conforms to the guidance of proposed GDC criterion 40 and criterion 42. The NRC staff reviewed the licensee's evaluation of environmental conditions in BFN Unit 1 and concludes that the BFN Unit 1 environmental conditions are enveloped by the GE equipment qualification parameters established for the NUMAC-PRNMS modification.

As described in NEDC-32410P-A, the PRNMS uses the same panel interfaces as the existing power range monitor equipment. High frequency filters are installed on the ac power supply, and shielded cables for all signal leads will be used in lieu of testing nonsafety equipment noise effects on the PRNMS.

The NRC staff finds the licensee's evaluation of the EMI environment and the measures taken to reduce adverse EMI affects to be an acceptable approach for ensuring the NUMAC-PRNMS EMI environment conforms to the guidance of proposed GDC criterion 40 and criterion 42 for protection against adverse environmental effects.

3.1.1.5 Administrative controls

In the SE of NEDC-32410P-A, the NRC staff found the NUMAC-PRNMS design features that control access to setpoint adjustments, calibrations, and test points acceptable. Since the licensee has not proposed design changes that would override these controls, the NRC staff finds that the licensee has acceptable controls for controlling access to the PRNMS panel and the APRM/OPRM channel bypass switch.

3.1.1.6 Confirmation of Human Factors Review

The licensee stated that the BFN Unit 1 design change process and implementing procedures require completion of a Human Factors Engineering (HFE) Process Checklist and performance of an HFE review of changes to the plant operator's panel. The licensee further stated that an HFE review, per applicable BFN Unit 1 procedures of the proposed changes to the operator's panel, will be performed, and documentation of that review will be included in the final design package(s) for the PRNMS. The NRC staff performed a control room design review as part of the BFN Unit 1 restart recovery efforts. The NRC staff selected a sample of high safety significant and verified that the licensee had developed modification design packages or other appropriate measures. No significant issues were identified. Therefore, based on the above and that the licensee previously performed these changes to Units 2 and 3 with no major challenges, the NRC staff finds this acceptable.

On the basis of the above review and justifications for TS changes, the NRC staff concludes that the licensee's proposed TS changes for BFN Unit 1 are consistent with the NRC staff-approved guidance in NEDC-32410P-A. The NRC staff further concludes that the licensee has properly addressed the plant-specific conditions described in the NRC staff's SE for NEDC-32410P-A, and, therefore, finds the NUMAC-PRNMS modification and associated TS changes to be acceptable.

3.1.2 ARTS/MELLLA Implementation

The function of the licensed allowable power/flow operating map is to define the normal operating condition of the reactor core used in determining the operating safety limits. The proposed TS change reflects operation of BFN Unit 1 in a region that is above the current rated rod line. The current approved operating envelope is modified to include the extended operating region bounded by the rod line that passes through the 100-percent power/99-percent core flow point shown in Figure 2-1 of Ref. 1. This extended operating domain is called the MELLLA. The analysis presented in Ref. 1 includes the MELLLA conditions and the equilibrium analysis performed for GE-14 fuel.

The function of the RBM is to prevent fuel damage in the event of erroneous rod withdrawal from locations of high-power density during high-power level operation. It does this by blocking control rod movement that could result in violating a thermal limit (the safety limit MCPR or the 1-percent cladding plastic strain limit) in the event of a Rod Withdrawal Error (RWE) event.

The functions of the APRM system include:

generation of a trip signal to scram the reactor during core-wide neutron flux transients before exceeding the safety analysis design basis,

blocking control rod withdrawal whenever operation exceeds set limits in the operating map, prior to approaching the scram level, and,

providing an indication of the core average power level in the power range.

The flow-biased rod block setdown and APRM flow-biased flux scram trip and alarm functions are provided to achieve these requirements.

The proposed partial implementation of the APRM, RBM Technical Specification (ARTS) improvement program will increase the plant operating efficiency by updating the thermal limits requirements to be consistent with current GE methodology and from improvements in plant instrumentation accuracy and response by incorporating digital flow control trip reference (FCTR) cards to replace the original analog cards.

The ARTS improvement program includes changes to the current APRM system, which requires the TS changes described in Section 3.11 below. The current BFN Unit 1 improved TSs require that the flow-referenced APRM scram and rod block trips be lowered (setdown) when the core Maximum Total Peaking Factor exceeds the design Total Peaking Factor. An alternative to an actual “setdown” is accomplished by adjusting the APRM gain upwards to achieve the desired equivalent result. The basis for the current APRM trip setdown requirement corresponds to the Hench-Levy Minimum Critical Heat Flux Ratio thermal limit criteria (Ref. 3). A subsequent update to the thermal limits requirements, which decreases the dependence on the local thermal hydraulic conditions, including the core peaking factors, was developed by GE. The resulting GE Thermal Analysis Basis critical power ratio correlation model (Ref. 3), which relies on bundle boiling length and exit quality, was reviewed and approved by the NRC staff.

The objective of the APRM improvements is to justify removal of the APRM trip setdown and the Design Total Peaking Factor requirement. Since the elimination of the APRM gain and setpoint requirement can potentially affect the fuel thermal-mechanical integrity and the ECCS LOCA performance, the NRC staff reviewed the acceptability of these changes. The following criteria, contained in NUREG-0800, Chapters 4 and 15, were used by the NRC staff to assure satisfaction of the applicable licensing requirements to demonstrate acceptability of the APRM trip setdown requirement:

1) The safety limit minimum critical power ratio (SLMCPR) shall not be violated as a result of any anticipated operational occurrences (AOOs),
2) All fuel thermal-mechanical design bases shall remain within the licensing limits described in the GE generic fuel licensing report (GESTAR-II), and,

3) The peak cladding temperature and the maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

The ARTS-specific changes are summarized here:

1. The requirement for setdown of the APRM scrams and rod blocks is deleted,
2. New power-dependent MCPR adjustment factors, MCPR(p), are added,
3. New flow-dependent MCPR adjustment factors, MCPR(f), replace the K_f multiplier,
4. New power-dependent and flow dependent maximum average planar linear heat generation rate (MAPLHGR) and,
5. The affected TS SRs, LCOs, and the associated Bases are modified or deleted, as required.

The NRC staff reviewed the safety analyses and systems response evaluations performed by General Electric Nuclear Energy (GENE) to justify BFN Unit 1 operation in the expanded MELLLA region, as submitted in Refs. 1 and 2. The GE-14 equilibrium analyses presented in Ref. 1 was specifically for Unit 1. However, the analysis presented in Ref. 2 was for all three units. The common analyses for BFN units fuel independent evaluations, such as containment response, were performed based on the current hardware design and applicable plant geometry for BFN Unit 1.

3.1.2.1 Method of Analysis

The analyses that are used to justify operation with the ARTS improvement and the MELLLA power/flow operating map are based on NRC-approved computer codes, methodologies, and applicable industry standards, which are discussed in the ARTS/MELLLA safety analysis report (A/MSAR) and associated references. Section 12 of Ref. 2 lists the nuclear steam supply system (NSSS) computer codes used in the safety analyses.

The ARTS thermal limits are expected to be fuel cycle-independent and the ARTS transient analyses in Ref. 2 were performed at the CLTP plant conditions for Unit 2 core configuration, using the GENE standard reload licensing methodology described in the GE Standard Application for Reactor Fuel, Rev. 14, dated June 2000 documentation. The BFN Unit 1 plant-specific evaluations will be performed to establish plant-unique, flow-dependent MCPR, LHGR, and MAPLHGR limits.

The NRC staff finds the methods used to be acceptable, since approved methods are used and the example calculations provided demonstrate the application of the methods to the proposed ARTS/MELLLA implementation at the CLTP condition.

3.1.2.2 Fuel Thermal Limits

The NRC staff reviewed the effects of operation along the higher MELLLA rod line at the CLTP on the thermal limits and the thermal limits management with the ARTS power and flow dependent limits, which are covered in the A/MSAR.

The potentially limiting AOOs and accident analyses were evaluated to support BFN Unit 1 EPU operation (with the ARTS off-rated limits, as well as operation in the MELLLA region for the reference equilibrium core of GE 14 (Table 9-2 of Ref. 1). The NRC staff approved evaluation methods presented in Ref. 8 were used for the transient analysis.

The following events were evaluated:

1. Generator Load Rejection with Bypass Failure;
2. Turbine Trip With Bypass Failure
3. Feedwater Controller Failure (FWCF) Maximum Demand
4. Feedwater Controller Failure with Turbine Bypass Out of Service (TBOOS)
5. Pressure Regulator Downscale Failure
6. Loss of Feedwater Heating
7. Inadvertent High Pressure Coolant Injection actuation
8. Rod Withdrawal Error
9. Slow Recirculation Increase
10. Fast Recirculation Increase
11. Generator Load Rejection with Bypass
12. Main Steam Isolation Valve (MSIV) Closure-All Valves
13. MSIV Closure-One Valve
14. Loss of Feedwater Flow
15. Loss of One Feedwater Pump

The most limiting transients are the following:

Generator Load Rejection with Bypass Failure
Turbine Trip With Bypass Failure
FWCF Maximum Demand
Feedwater Controller Failure maximum demand with TBOOS

Extensive transient analyses at a variety of power and flow conditions, performed for the original ARTS improvement program, established a database of limiting transients, which were representative of a variety of plant configurations and key parameters designed to be applicable to all BWRs. These generic evaluations determined power-dependent trends for two operating ranges. The first range is between the 100-percent rated thermal power and the power level [30 percent of CLTP for BFN Unit 1] where the reactor scram on turbine stop valve closure or turbine control valve fast closure is bypassed (Pbypass). The second range is from Pbypass (30 percent) down to 25 percent of the CLTP. The current BFN Unit 1 TS does not require thermal limit monitoring below 25 percent of CLTP.

BFN Unit 1 specific analyses will be performed to confirm the applicability of the generic power-dependent limit multipliers [K(p), and MAPLHGR(p)] above Pbypass. BFN Unit 1 specific evaluations will also be performed between Pbypass (30 percent) and 25-percent power to establish BFN Unit 1 unique MCPR, LHGR, and MAPLHGR limits, which will apply to future reloads of GE fuel designs through the GE 14 design. BFN Unit 1 specific evaluations will also be performed to establish the flow-dependent MCPR and MAPLHGR limits.

The MELLLA operating conditions have only a small effect on the operating limit maximum critical power ratio (OLMCPR). The OLMCPR is calculated by adding the change in MCPR due to the limiting AOO event to the SLMCPR and will be determined with the actual core analysis before start-up. Analyses were performed for BFN Unit 1 before startup and beyond to provide a BFN Unit 1 plant-specific, statistically-based, power-dependent RWE OLMCPR value.

The cycle-specific reload fuel analyses will determine the limits for rated and applicable off-rated conditions, and application of the methodology is demonstrated by the analyses performed for the EPU operating cycle. Therefore, the NRC staff finds this approach acceptable.

3.1.2.3 Vessel Overpressure Protection

The MSIV closure with a flux scram event is used to determine compliance to the industry standard American Society of Mechanical Engineers Pressure Vessel Code. A plant-specific calculation was performed for a representative core at 102 percent of EPU power level (Ref. 1). The peak calculated pressure for the limiting event was 1314 pounds per square inch gage (psig) less than the accepted criterion of 1375 psig and, hence, is acceptable. NRC staff approved evaluation model Ref. 8 was used for the analyses.

Because the licensee will perform cycle specific overpressure evaluations before start-up, using approved methods, the NRC staff finds this acceptable.

3.1.2.4 Thermal-Hydraulic Stability

This change will be addressed in the NRC staff review of the January 6, 2006, submittal, and are not approved in this amendment.

3.1.2.5 LOCA Analysis

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. The MAPLHGR operating limit is based on the most limiting LOCA and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46.

For most BWRs, the full-spectrum ECCS-LOCA analyses were performed when the plants converted to the SAFER/GESTR analysis. For BFN Unit 1, this equilibrium analysis (Ref. 1) was performed based on operation along the current ELLLA rod line up 120 percent to the original licensed thermal power.

The current basis also specifies the APRM setdown requirement of a maximum LHGR value as a function of drive flow. This requirement is proposed to be replaced with the direct core power and flow dependent fuel thermal limits of the ARTS improvement program, which is not required for the LOCA analysis. The NRC staff reviewed the current licensing basis analysis to determine the ECCS performance effect of operation in the MELLLA domain. An evaluation was performed with GE 14 and GE 13 fuel to demonstrate compliance with the ECCS-LOCA acceptance criteria.

The NRC staff approved evaluation methods for GE Reports NEDC-32484P, NEDE-23785P-A, and NEDO-20566A (Refs. 4, 5, and 6) that were used for the LOCA analyses.

For MELLLA/EPU operation, a large break peak cladding temperature (PCT) value of 1830 °F was determined for GE 14 fuel and 1780 °F for GE 13 fuel. This provides an adequate margin to the 2200 °F PCT limit. Justification for the elimination of the former 1600 °F Upper Bound PCT limit was provided in the GE 14 BFN Unit 1 analysis. Also, the maximum local oxidation is less than 3 percent, and the core-wide metal-water reaction is less than 0.1 percent. Because the licensee performed a representative ECCS-LOCA analysis using 10 CFR 50.46 and Appendix K requirements, MELLLA operation is acceptable.

The NRC staff has determined that no additional operating restrictions would be required for ARTS/MELLLA operation, since the determination of the sensitivity of the ECCS-LOCA evaluations to operation in the MELLLA domain shows compliance with the acceptance criteria.

3.1.2.6 Anticipated Transient without Scram (ATWS)

The NRC staff reviewed the BFN Unit 1 representative analysis that was performed using the approved licensing methodology (Ref. 1) to demonstrate compliance with 10 CFR 50.62 ATWS requirements. The analysis has been performed for EPU RTP. The loss of off-site power and inadvertent opening of a relief valve events were determined to be nonlimiting.

The NRC staff concludes, based on its review of the ATWS analyses described above, that BFN Unit 1 meets the ATWS mitigating features stipulated in 10 CFR 50.62 and that the results of the ATWS analyses for MELLLA operation at the EPU RTP meet the ATWS acceptance criteria. This approach is acceptable because the ATWS analyses at EPU RTP bound the ATWS analyses at CLTP.

3.1.2.7 Testing

Standard pre-operational testing (APRM, recirculation flow calibrations) will be performed after installation of the new digital FCTR cards. The APRMs will be calibrated prior to MELLLA implementation. The APRM flow-biased scram and rod block setpoints will be consistent with the ARTS/MELLLA implementation, with all APRM trips and alarms tested. The flow-biased setpoints for the RBM will be confirmed. The NRC staff finds this acceptable.

The NRC staff has reviewed the licensee’s license amendment request application along with the supporting documentation, including responses from the request for additional information.

The review of TS changes in this SE is performed solely to evaluate the changes that would be required to support the ARTS/MELLLA implementation at BFN Unit 1. This review covered the ARTS/MELLLA application with a representative GE 14 analysis at EPU. Currently, Unit 1 is defueled and is scheduled to restart in 2007. TVA stated that the fuel supplier for Unit 1 will be GNF/GE. The bases for acceptance of MELLLA operation are approved GE evaluation methods and the use of only GE fuel for Unit 1.

Information provided by TVA as part of the EPU application dated June 28, 2004, related to the MELLLA was used in support of this review. The NRC staff finds this approach acceptable because MELLLA conditions at EPU bound the MELLLA conditions at CLTP. Approval of this amendment does not constitute authority to operate above the CLTP.

Based on its review, the NRC staff concludes that the proposed TS changes are acceptable because the safety analysis supporting actual operation in the ARTS/MELLLA regimes have been reviewed and the NRC staff concludes that operation will not endanger the public health and safety.

3.1.3 Plant-Specific Revised TSs

The NRC staff reviewed the proposed changes to the BFN Unit 1 TSs that are identified in the licensee's submittal. The NRC staff review was focused on MELLLA operation and not the EPU operation. The approval of these TS sections are only for MELLLA conditions and not for EPU. The changes include deletion of the current setdown requirement, and new power and flow-dependent MCPR and MAPLHGR limits. The proposed TS changes are as follows:

3.1.3.1 Change 1 - Page I, Table of Contents

The Table of Contents listing of Section 3.2.4, APRM Gain and Setpoints, has been deleted. This is a conforming editorial change related to change 3, discussed below. Therefore, the NRC staff finds this change acceptable.

3.1.3.2 Change 2 - Page 1.1-5, Section 1.1

The definition of "Maximum Fraction" of limiting power density (MFLPD) is deleted. The definition of MFLPD is no longer required in the TSs due to the ARTS/MELLLA implementation. Therefore, the NRC staff finds this change acceptable.

3.1.3.3 Change 3 - Pages 3.2-7 and 3.2-8, Section 3.2.4

Section 3.2.4, the LCO and SR entitled "Average Power Range Monitor (APRM) Gain and Setpoints," is deleted in its entirety. Section 3.2.4 is no longer required in the TSs due to the ARTS/MELLLA implementation. Therefore, the NRC staff finds this change acceptable.

3.1.3.4 Change 4 - Table 3.3.1.1-1, Reactor Protection System Instrumentation

In Change 4, the licensee proposed adding a Note for Required Action A.2 and for Condition B. The Note states, "Not applicable for Functions 2.a, 2.b, 2.c or 2.d." The licensee stated that neither Required Action A.2 nor Condition B is applicable for APRM Functions 2.a, 2.b, 2.c, or 2.d. Required Action A.2 is not applicable for Functions 2.a, 2.b, 2.c, or 2.d because in the new configuration, inoperability of one APRM channel affects both RPS trip systems. Thus, with an inoperable APRM channel, Required Action A.1 must be satisfied and is the only action (other than restoring operability) that will restore capability to accommodate a single failure.

Condition B also is not applicable because inoperability of more than one required APRM channel results in loss of trip capability; thus, in this circumstance entry is required into Condition C, as well as into Condition A for each channel. The licensee's justifications for these two changes are consistent with the APRM 2-out-of-4 voter function described in NEDC-32410P-A, its associated operability requirements, notes, operating modes, and action statements. The changes are consistent with NEDC-32410P-A and related topical reports. Therefore, the NRC staff finds these changes acceptable.

3.1.3.5 Change 5 - Page 3.3-3, Section 3.3.1.1

SR 3.3.1.1.2 is revised to delete the APRM gain adjustment required by LCO 3.2.4. The deleted portion is no longer required in the TSs due to the ARTS/MELLLA implementation. Therefore, the NRC staff finds this change acceptable.

3.1.3.6 Change 6 - Table 3.3.1.1-1, Reactor Protection System Instrumentation

In Change 6, the licensee proposed, in the table of SRs for LCO 3.3.1.1, revising the note in SR 3.3.1.1.9 to remove the APRM Function 2.a reference, which is now addressed in SR 3.3.1.1.13. This change is consistent with the TS changes recommended and approved by the NRC staff in NEDC-32410P-A. Therefore, the NRC staff finds this change acceptable.

3.1.3.7 Change 7 - Table 3.3.1.1-1, Reactor Protection System Instrumentation

In Change 7, the licensee proposed deleting SR 3.3.1.1.11 and the corresponding surveillance frequency. This proposed change reflected the inclusion of the recirculation flow loop calibrations as part of the overall Channel Calibration (SR 3.3.1.1.13) for APRM Function 2.b. SR 3.3.1.1.11 required a calibrated flow signal be used to verify the accuracy of the total loop drive flow signal for the APRM Flow Biased Simulated Thermal Power - High function. Since calibration is proposed according to SR 3.3.1.1.13, deleting SR 3.3.1.1.11 is appropriate. The NRC staff reviewed the proposed change and found it appropriate for BFN Unit 1 because it is consistent with NEDC-32410P-A. Therefore, the NRC staff finds this change acceptable.

3.1.3.8 Change 8 - Table 3.3.1.1-1, Reactor Protection System Instrumentation

In Change 8, the licensee proposed adding a new note to SR 3.3.1.1.13 that excludes the neutron detectors from the channel calibration. As discussed in the TS Bases section, the neutron detectors are excluded from Channel Calibration because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. The NRC staff reviewed the proposed change and found it appropriate for BFN Unit 1 because it is consistent with NEDC-32410P-A. Therefore, the NRC staff finds this change acceptable.

3.1.3.9 Change 9 - Table 3.3.1.1-1, Reactor Protection System Instrumentation

In Change 9, the licensee proposed adding a new Channel Functional Test (SR 3.3.1.1.16) with a 184-day frequency. The licensee also proposed an accompanying note allowing 12 hours to complete the requirement for APRM Function 2.a when entering MODE 2 from MODE 1. The licensee stated that testing of APRM Function 2.a cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links; consequently, the note is required to provide 12-hours in which to bring current to the Channel Calibration for APRM Function 2.a when entering MODE 2 from MODE 1. The 12 hour time requirement was based on operating experience and consideration of providing a reasonable time in which to complete the SR. The NRC staff reviewed the proposed change and found it appropriate for BFN Unit 1 because it is consistent with NEDC-32410P-A. Therefore, the NRC staff finds this change acceptable.

3.1.3.10 Change 10 - Table 3.3.1.1-1, Reactor Protection System Instrumentation

In Change 10, the licensee proposed five groups of changes (A, B, C, D, E) to LCO 3.3.1.1 and Table 3.3.1.1-1.

3.1.3.10.1 Change A. The licensee proposed changing from 2 to 3 the required minimum number of operable instrument channels per trip system for the APRM high and inoperable scram trip functions because the new configuration will have four total APRM channels combined in a 2-out-of-4 logic. In the proposed configuration, a minimum of three of the four channels will be required operable to meet single failure criteria for the RPS trips initiated by APRMs. Additionally, the licensee proposed adding Note “b" to Table 3.3.1.1-1 to highlight that, in the new configuration, each APRM instrument channel will provide input to both RPS trip systems. These two changes were approved by the NRC staff in NEDC-32410P-A, therefore, the proposed changes are acceptable.

3.1.3.10.2 Change B. The licensee proposed deleting the requirement for an APRM downscale scram trip function. The licensee stated that the APRM downscale scram trip is not credited with performing any safety function. Deletion of this APRM function was approved by the NRC staff in NEDC-32410P-A, therefore, the proposed change is acceptable.

3.1.3.10.3 Change C. The licensee proposed adding a new 2-out-of-4 voter function, with a minimum of two operable instrument channels per RPS trip system. This requirement is consistent with the NUMAC-PRNMS hardware configuration. There are two voters per RPS trip system, and requiring two voters operable in each of the two RPS trip systems ensures that the single failure criteria is met. Because operability of the voters is required whenever any other APRM trip function is required, the applicable modes for voter operability are MODE 1 and MODE 2. Inoperability of one or more voters will result in entry to Condition A, Condition B, or Condition C, as appropriate. Failure to complete the required actions within the allowable completion times requires that the reactor be in MODE 3 (where APRM operability is not required) within 12 hours. These changes are consistent with the changes that were approved by the NRC staff in NEDC-32410P-A, therefore, the proposed changes are acceptable.

3.1.3.10.4 Change D. The licensee proposed the following changes to the SRs for APRM Functions 2.a, 2.b, 2.c, and 2.d (previously numbered “2.e”), and the addition of surveillance requirements for the new APRM Function 2.e (2-out-of-4 voter). Specific discussions for each of the five functions are provided in the following sections.

3.1.3.10.4.1 APRM Function 2.a, Neutron Flux - High (Setdown). For APRM Function for APRM Function 2.a, Neutron Flux - High, (Setdown), the licensee proposed the following changes:

- Delete the Channel Functional Test (SR 3.3.1.1.3) with 7-day frequency and, in its place a Channel Functional Test (SR 3.3.1.1.16) with a 184-day frequency is added,
- Replace the 92-day Channel Calibration (SR 3.3.1.1.9) and the corresponding reference in SR 3.3.1.1.9 with the SR 3.3.1.1.13 Channel Calibration and an 18-month frequency, and
- Delete the 18-month Logic System Functional Test (SR 3.3.1.1.14).

These changes are supported by the reliability analysis presented in NEDC-32410P-A, and are consistent with the recommendations of NEDC-32410P-A. Therefore, the proposed changes are acceptable.

3.1.3.10.4.2 APRM Function 2.b, Flow Biased Simulated Thermal Power - High. For APRM Function 2.b, Flow Biased Simulated Thermal Power - High, the licensee proposed the following changes:

- Replace the 92-day Channel Functional Test (SR 3.3.1.1.8) with the 184-day Channel Functional Test (SR 3.3.1.1.16),
- Replace the 92-day Channel Calibration (SR 3.3.1.1.9) and the corresponding reference in SR 3.3.1.1.9 with the 18-month Channel Calibration (SR 3.3.1.1.13),
- Include the 18-month flow signal calibration (SR 3.3.1.1.11) as part of the 18-month SR 3.3.1.1.13, and
- Delete the 18-month Logic System Functional Test (SR 3.3.1.1.14).

These changes in testing and surveillance are supported by the reliability analysis presented in NEDC-32410P-A, and are consistent with the recommendations of NEDC-32410P-A. Therefore, the proposed changes are acceptable.

3.1.3.10.4.3 APRM Function 2.c, Neutron Flux - High. For APRM Function 2.c, Neutron Flux - High, the licensee proposed the following changes:

- Replace the 92-day Channel Functional Test (SR 3.3.1.1.8) with a 184-day Channel Functional Test (SR 3.3.1.1.16),
- Replace the 92-day Channel Calibration (SR 3.3.1.1.9) and the corresponding reference in SR 3.3.1.1.9 with an 18-month Channel Calibration (SR 3.3.1.1.13), and
- Delete the 18-month Logic System Functional Test (SR 3.3.1.1.14).

These changes in testing and surveillance are supported by the reliability analysis presented in NEDC-32410P-A, and are consistent with the recommendations of NEDC-32410P-A. Therefore, the proposed changes are acceptable.

3.1.3.10.4.4 APRM Function 2.d, Inop. For APRM Function 2.d, Inop, the licensee proposed the following changes:

- Delete the calibration of local power range monitors (SR 3.3.1.1.7). This calibration remains a requirement of APRM Functions 2.a, 2.b and 2.c, where the local power range monitors provide direct inputs to the process signals monitored by the APRM trip functions,
- Replace the 92-day Channel Functional Test (SR 3.3.1.1.8) with a 184-day Channel Functional Test (SR 3.3.1.1.16), and
- Delete the 18-month Logic System Functional Test (SR 3.3.1.1.14).

These changes in testing and surveillance are supported by the reliability analysis presented in NEDC-32410P-A, and are consistent with the recommendations of NEDC-32410P-A. Therefore, the proposed changes are acceptable.

3.1.3.10.4.5 APRM Function 2.e, 2-out-of-4 voter. For the new APRM Function 2.e, 2-out-of-4 voter, the licensee proposed the following additions to the TSs:

- A 24-hour Channel Check (SR 3.3.1.1.1) consistent with the Channel Check frequency for the other APRM Functions,
- An 18-month Logic System Functional Test (SR 3.3.1.1.14), and
- A 184-day Channel Functional Test (SR 3.3.1.1.16).

These changes in testing and surveillance are supported by the reliability analysis presented in NEDC-32410P-A, and are consistent with the recommendations of NEDC-32410P-A. Therefore, the proposed changes are acceptable.

3.1.3.10.5 Change E. The licensee proposed changing the Allowable Value of APRM Function 2.b., Flow Biased Simulated Thermal Power-High

The licensee proposed to change the flow biased simulated thermal power-high from $\leq$ 0.58W + 62% RTP and $\leq$120% RTP to $\leq$ 0.66W + 71% RTP and $\leq$ 120% RTP. The allowable value for single loop operation is similarly changed. The flow-biased APRM scram setpoint maximum (clamped) allowable value of 120% does not change.

These changes incorporate new setpoints for the flow-biased APRM scram and rod block functions based on the MELLLA analytical limits. Therefore, the NRC staff finds these changes acceptable.

3.1.3.11 Change 11 - LCO 3.3.2.1 Control Rod Block Instrumentation SR 3.3.2.1.1 (Rod Block Monitor Channel Functional Test)

The licensee proposed changing the frequency of SR 3.3.2.1.1 from 92 days to 184 days. This change is supported by the reliability analysis presented in NEDC-32410P-A, and is consistent with the recommendations of NEDC-32410P-A. Therefore, the NRC staff finds this change acceptable.

3.1.3.12 Change 12 - LCO 3.3.2.1 Control Rod Block Instrumentation SR 3.3.2.1.4 (Rod Block Monitor Channel Calibration)

The licensee proposed changing the frequency of SR 3.3.2.1.4 from 92 days to 18 months. This change is supported by the reliability analysis presented in NEDC-32410P-A, and is consistent with the recommendations of NEDC-32410P-A. Therefore, the NRC staff finds this change acceptable.

3.1.3.13 Change 13 - Page 3.3-19, Section 3.3.2.1

A new SR Section 3.3.2.1.8 is added to support the change to power-dependent RBM setpoints. This SR has the same requirements as SR 3.3.2.1.4 in the BWR/4 ISTSs, NUREG-1433, Rev. 3. Changes in fuel types or licensed thermal power are not being addressed by this proposed change. Therefore, the NRC staff finds this acceptable because this change is consistent with NUREG-1433, Rev. 3.

3.1.3.14 Change 14 - Page 3.3-20, Table 3.3.2.2-1

The RBM portion of Table 3.3.2.1-1 and associated notes are revised to reflect the change from flow-biased to power-dependent RBM limits. These proposed changes are consistent with

Table 3.3.2.1-1 of BWR/4 ISTS NUREG-1433, Rev. 3. Changes in fuel types or licensed thermal power are not being addressed by this proposed change. Therefore, the NRC staff finds this acceptable because this change is consistent with NUREG-1433, Rev. 3.

3.1.3.15 Changes 15 and 16

The licensee proposed to modify Figure 3.4.1-1, Thermal Power versus Core Flow Stability Regions, to expand Region II to include the power/flow map segment between 45 percent and 50 percent core flow. SR 3.4.1.2 was also proposed to be modified to match the requested change to figure 3.4.1-1. These changes will be addressed in the NRC staff review of the January 6, 2006, submittal, and are not approved in this amendment.

3.1.3.16 Changes 17 (LCO 3.10.8.a) and 18 (SR 3.1.8.1), Shutdown Margin Test-Refueling

Reference to function 2.d (now the "Inop" function) is added for consistency with the previous changes to Table 3.3.1.1-1. Reference 2.e in LCO 3.10.8.a and SR 3.10.8.1 is changed to correspond to the new 2-out-of-4 voter function in table 3.3.1.1-1. These changes are editorial, therefore, the NRC staff finds them acceptable.

3.1.3.17 Bases Changes

The licensee also provided the associated TS Bases that reflect the proposed TS changes as an attachment to its application. The TS Bases changes are consistent with the licensee's proposed plant-specific TS changes, and the NRC staff has no objections to the Bases changes presented in the licensee's application. The changes related to the activation of thermal-hydraulic stability monitoring instrumentation will be addressed in the NRC staff review of the January 6, 2006, submittal, and are not approved in this amendment.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff attempted to contact the Alabama State official regarding the proposed issuance of the amendment. There was no official response.

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 and changes SRs. 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 (69 FR 5208). 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.

7.0 REFERENCES

1. Letter from Abney, T. E., TVA, to USNRC, “BFN1-Unit 1-Proposed TS Change 431-Request for License Amendment-EPU Operation,” TVA-BFN1-TS-431, dated June 28, 2004. (Attachment: “Browns Ferry Unit 1 Safety Analysis Report for Extended Power Uprate," NEDC-33101P, GE Nuclear Energy, June 2004.)

2. “Maximum Extended Load Line Limit And Arts Improvement Program Analyses (ARTS/MELLLA), Browns Ferry Nuclear Plant Unit 1, 2, 3," NEDC-32433P, GE Nuclear Energy, April 1995.

3. “General Electric Standard Application for Reactor Fuel,” GE Nuclear Energy, NEDE-24011-P-A-14 and NEDE-24011-P-A-14-US, Revision 14, June 2000.

4. NEDC-32484P, “BFN 1, 2, 3 SAFER/GESTR-LOCA Analysis,” January 2002.

5. NEDE-23785P-A, Vol. III, Supplement 1, Revision 1, “GESTR-LOCA and SAFER Models for Evaluation of Loss-of-Coolant Accident Volume III, Supplement 1, Additional Information for Upper Bound PCT Calculation,” March 2002.

6. NEDO-20566A, “General Electric Model for LOCA analysis in Accordance with 10 CFR 50 Appendix K,” September 1986.

7. Browns Ferry Nuclear plant Updated Final Safety Analysis Report (UFSAR).

8. NEDC-24154P-A, “Qualification of the One Dimensional Core Transient Model (ODYN) for Boiling Water Reactors (Supplement 1 - Volume 4),” February 2000.

9. NEDC-32523P-A, “ Generic Evaluation of GE BWR EPU," February 2000.

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