April 25, 2005

TVA-BFN-TS-431

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN, P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	

BROWNS FERRY NUCLEAR PLANT (BFN) – UNIT 1 - RESPONSE TO NRC's REQUEST FOR ADDITIONAL INFORMATION RELATED TO TECHNICAL SPECIFICATIONS (TS) CHANGE NO. TS-431– REQUEST FOR EXTENDED POWER UPRATE OPERATION (TAC NO. MC3812)

This letter contains the additional information requested by the NRC Staff concerning testing at EPU conditions. TS-431 which was submitted on June 28, 2004 (Reference 1), requested a license amendment and TS changes that support a requested increase in the reactor thermal power level to 3952 MWt, an approximate 20 percent increase in thermal power.

By letter dated February 23, 2005 (Reference 2), TVA supplemented the application, providing additional information requested by the NRC on Large Transient Testing. In Enclosure 8 of Reference 1, and in Reply 4 of Reference 2, TVA provided justification for elimination of Large Transient Testing. However, based on February 10, 2005, and February 17, 2005, teleconferences between TVA management and NRC Staff, it was determined that the justification for the elimination of Large Transient Testing would be in accordance with NUREG-0800, Standard Review Plan (SRP), Section 14.2.1, Draft, Revision 0, “Generic Guidelines for Extended Power Uprate Testing Programs.”

The Enclosure to this letter provides TVA's BFN Extended Power Uprate Testing Program and further justification for elimination of large transient testing. The enclosure replaces Enclosure 8 of the initial BFN Unit 1 application and the information in Reply 4 of the February 23, 2005, letter in their entirety. The information has been revised, expanded and in some cases reordered to fully address the review criteria contained in SRP Section 14.2.1. The basic content contained in Table 1 of this letter was provided in Enclosure 8 of Reference 1, but has been revised substantially, and renumbered for clarification. In the initial submittal, if a test was not planned specifically to address EPU implementation, the "Testing Planned for EPU" column was marked as "No." In Table 1 of this enclosure, where testing such as normal startup testing required by the BFN Technical Specifications would perform either a full or partial startup test, Table 1 contains a "Yes" in the column with an associated explanation. Therefore, some of the responses for the column entitled "Testing Planned for EPU", have been changed since our initial submittal.

Table 2 provides a comparison of the steady state and transient tests from the initial BFN startup tests to those described in SRP 14.2.1, Attachments 1 and 2. Table 3 tabulates the modifications, setpoint adjustments, and parameter changes required to implement EPU, and identifies the EPU tests associated with those changes, addressing the review criteria contained in SRP 14.2, Section III.B.

TVA is providing similar information regarding the Units 2 and 3 EPU application in a separate submittal. There are no new regulatory commitments associated with this submittal. If you have any questions concerning this letter, please telephone me at (256) 729-2636.

Pursuant to 28 U.S.G. § 1796 (1994), I declare under penalty of perjury that the forgoing is true and correct. Executed on this 25th day of April, 2005.

Sincerely,

Original signed by:

T. E. Abney
Manager of Licensing
and Industry Affairs

References:

1. TVA letter, T. E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) – Unit 1 - Proposed Technical Specifications (TS) Change TS - 431- Request for License Amendment - Extended Power Uprate (EPU) Operation," dated June 28, 2004.

2. TVA letter, T. E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) – Unit 1 Response to NRC's Acceptance Review Letter and Request for Additional Information Related to Technical Specifications (TS) Change No. TS-418, Request for Extended Power Uprate Operation," dated February 23, 2005.

Enclosure
cc (Enclosure):
(Via NRC Electronic Distribution)

U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-3415

Mr. Stephen J. Cahill, Branch Chief
U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

NRC Senior Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, AL 35611-6970

Margaret Chernoff, Senior Project Manager
U.S. Nuclear Regulatory Commission
(MS 08G9)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Eva A. Brown, Project Manager
U.S. Nuclear Regulatory Commission
(MS 08G9)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

MJB:BAB
Enclosure
cc (w/o Enclosure):

A. S. Bhatnagar, LP 6A-C
J. C. Fornicola, LP 6A-C
R. G. Jones, NAB 1A-BFN
K. L. Krueger, POB 2C-BFN
R. F. Marks, PAB 1C-BFN
F. C. Mashburn, BR 4X-C
N. M. Moon, LP 6A-C
J. R. Rupert, NAB 1A-BFN
K. W. Singer, LP 6A-C
M. D. Skaggs, PAB 1E-BFN
E. J. Vigluicci, ET 11A-K
NSRB Support, LP 5M-C
EDMS WT CA - K (with Enclosure)

s:lic/submit/Techspec/BFN Units 1 EPU Testing Program TS 431.doc

ENCLOSURE 1

**TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT UNIT 1
DOCKET NO. 50-259**

RESPONSE TO NRC's REQUEST FOR ADDITIONAL INFORMATION RELATED TO TECHNICAL SPECIFICATIONS (TS) CHANGE NO. TS-431– REQUEST FOR EXTENDED POWER UPRATE OPERATION

I. INTRODUCTION

By letter dated June 28, 2004 (Reference 1), TVA submitted to the NRC a license amendment application requesting authorization for Extended Power Uprate (EPU) operation for Browns Ferry Nuclear Plant (BFN) Unit 1. By letter dated February 23, 2005 (Reference 2), TVA supplemented that application, providing additional information requested by the NRC as part of their acceptance review. In Enclosure 8 of the initial application, and in Reply 4 of the February 23, 2005 supplemental submittal, TVA provided justification for elimination of the requirement for large transient testing upon implementation of the EPU. The initial amendment application provided an EPU safety analysis report for EPU herein referred to as the PUSAR.

This submittal replaces Enclosure 8 of Reference 1 in its entirety. The information has been revised, expanded and in some cases reordered to fully address the review criteria contained in the NUREG-0800 (the Standard Review Plan or “SRP”), Section 14.2.1, Draft, Revision 0. The basic content contained in Tables 1 and 2 of this enclosure was provided in Enclosure 8 of Reference 1, but has been revised substantially, and renumbered for clarification. In our initial submittal, if a startup test was not planned specifically to address EPU implementation, the “Testing Planned for EPU” column was marked as “No.” In Table 1 of this enclosure, where testing such as normal startup testing required by the BFN Technical Specifications would perform either the full or partial initial startup test, Table 1 contains a “Yes” in the column with an associated explanation. Therefore, some of the responses for the column entitled “Testing Planned for EPU” have been changed since our initial submittal.

In accordance with SRP Section 14.2.1, this information demonstrates that structures, systems, and components (SSCs) will perform satisfactorily at the requested power level and thus the plant can be operated safely at the uprated power level.

II. BACKGROUND

BFN Unit 1 was voluntarily shutdown in 1985 and is in the process of being recovered to an operating status. Unit 1 original and current licensed reactor thermal power is 3293 MWt with a steam dome pressure of 1005 psig. As part of the recovery process, TVA will uprate the unit to 3952 MWt with an associated 30 psi pressure increase. The scope of modifications required to recover the unit will also include the modifications required to achieve the equivalent of the 105% uprate completed on Units 2 and 3 as well as those for 120% uprate. Thus, Unit 1 will be recovered in a manner that will allow it to operate within the same design/licensing basis as BFN Units 2 and 3 following implementation of 120% EPU on those units. As part of the BFN Unit 1 restart project, and in addition to the acceptance testing planned for the modifications needed to implement EPU, BFN Unit 1 will undergo extensive testing prior to and during power ascension as part of its Restart Test Program (RTP). The RTP is a comprehensive program developed to confirm that the plant is capable of meeting its safe shutdown requirements. Further information concerning the BFN Unit 1 RTP is provided in Section III.A below.

BFN Units 2 and 3 have been licensed to operate at 105% of the original licensed thermal power (OLTP), 3458 MWt. Unit 3 has been operating at these uprated conditions since the fall of 1998 and Unit 2 since the spring of 1999. The power uprate to 105% for Units 2 and 3 included a 30 psig vessel steam dome pressure increase. Because the steam dome pressure increase has already been accomplished, Units 2 and 3 will operate at the present reactor vessel pressure following implementation of EPU.

III. EPU TESTING PROGRAM REVIEW

The following sections address the review criteria of Standard Review Plan (SRP), Section 14.2.1, Subsection III, entitled "Review Procedures." The below sections III.A through III.D correspond to Sections in the SRP.

A. Comparison of Proposed EPU Test Program to the Initial Plant Test Program

1. General Discussion

A comparison of the proposed EPU testing program to the original startup test program performed during initial plant licensing is provided in Table 1 for all three units. The original BFN startup tests are included regardless of the power level at which they were performed.

The planned power ascension testing at BFN following uprate will provide a controlled and systematic testing program for the power levels above Current Licensed Thermal Power (CLTP) to EPU. Testing will be performed in accordance with the Technical Specifications (TS) and/or applicable procedures on instrumentation re-calibrated to EPU conditions.

Steady-state data will be taken during power ascension and continuing at each EPU power increase increment. This data will allow system performance parameters to be projected through the EPU power ascension. EPU power increases above 100% CLTP are planned along an established flow control/rod line in increments of equal to or less than 5% power. Steady-state and transient data, including fuel thermal margin, will be taken at each step. Routine measurements of reactor and system parameters will be evaluated, prior to the next power increment. Plant procedures will identify specifically planned EPU tests, the associated acceptance criteria and the appropriate test conditions. All testing will be done in accordance with written procedures as required by 10 CFR 50, Appendix B, Criterion XI.

Table 1 also provides the resulting EPU Power Ascension Test Plan matrix. The matrix specifies expected steady state and transient tests at different power levels. Similar matrices for the initial startup testing are contained in UFSAR Tables 13.5-4, 13.5-5, and 13.5-6.

In addition to the testing planned under the EPU Testing Program, BFN Unit 1 will perform a comprehensive Restart Test Program (RTP) as part of its restart program. The RTP is a component of the BFN Nuclear Performance Plan (NPP), which was submitted to NRC in 1986 to address TVA actions needed to restart all three BFN units. The NPP formed the basis for restart of Units 2 and 3 in 1991 and 1995 respectively. Likewise, the Unit 1 RTP forms part of the basis for the restart of BFN Unit 1. The BFN Unit 1 RTP is being developed based on the RTP developed, reviewed by the NRC, and approved for BFN Unit 3 as described and documented in NRC Safety Evaluation transmitted to TVA by letter dated August 30, 1994 (Reference 3). Lessons learned during implementation of both the BFN Units 2 and 3 RTPs are being incorporated into the development of the BFN Unit 1 RTP. In a December 13, 2002 letter (Reference 4), TVA transmitted its proposed regulatory framework for the restart of BFN Unit 1, which reiterated the intent to follow the Unit 3 precedent with regard to the restart test program. The NRC accepted the approach in its August 14, 2003, letter to TVA (Reference 5).

The RTP for Units 1 and 3 was submitted to NRC on September 27, 1991, (Reference 6), and supplemented on February 18, 1992 (Reference 7), December 28, 1992, (Reference 8) and July 19, 1993 (Reference 9). As a framework for its evaluation, the NRC reviewed the BFN RTP against the guidance contained in NRC Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants", Revision 2. As part of that review, TVA described differences between the BFN RTP and the preoperational test guidance contained in NRC Regulatory Guide 1.68. As a result of that review, the NRC concluded that implementation of the Browns Ferry RTP would ensure proper verification of the functional integrity of the safety systems at BFN.

NRC approved the RTP for Unit 3 by SER and letter dated August 30, 1994 (Reference 3). TVA is following the RTP program for Unit 1 as previously submitted and as reiterated in its December 13, 2002, letter to NRC transmitting TVA's proposed regulatory framework for the restart of Unit 1 (Reference 4), and accepted by the NRC Staff by letter dated August 14, 2003 (Reference 5).

A significant portion of the BFN Unit 1 RTP was derived from a comprehensive Safe Shutdown Analysis (SSA), which addresses all three units. The SSA is a systematic analysis of the requirements for ensuring the safe shutdown of BFN Units 1, 2, and 3 for transients, accidents, and special events. It documents the safety system actions for which credit has been taken in the UFSAR, reload analyses, and other communications concerning transients, accidents and special events. The events considered in the SSA included:

- Generator Trip
- Turbine-Generator Trip with Bypass Failure
- Pressure Regulator Failure – Closed
- Turbine Trip
- Isolation of all Main Steam Lines
- Closure of One Main Steam Isolation Valve
- Loss of Condenser Vacuum
- Loss of Feedwater Heater
- Shutdown Cooling (RHRS) Malfunction (Temperature Decrease)
- Inadvertent Pump Start
- Control Rod Withdrawal Error
- Fuel Assembly Insertion
- Control Rod Removal Error
- Pressure Regulator Failure – Open
- Inadvertent Opening of All Bypass Valves
- Inadvertent Opening of A Safety/Relief Valve
- Loss of Feedwater Flow
- Loss of Offsite AC Power
- Recirculation Control Failure-Decrease
- Recirculation Pump Trip (One Pump Trip)
- Recirculation Pump Trip (Two Pump Trip)
- Recirculation Pump Seizure
- Recirculation Flow Controller Failure Increasing Flow
- Startup of Idle Reactor Recirculation Pump
- Loss of Shutdown Cooling
- Feedwater Controller Failure – Maximum Demand
- Control Rod Drop Accident
- Pipe Break Inside Containment – Large Break
- Intermediate Pipe Break Inside Containment
- Small Pipe Break Inside Containment
- Pipe Break Inside – Containment and Radiological Effect
- Fuel Handling Accident
- Pipe Break Outside Containment
- Shutdown From Outside of Control Room
- Shutdown Without Control Rods
- Overpressure Protection (MSIV Closure – Backup Scram)
- Rotated or Mis-located Bundle
- Flood
- Low Reservoir Downstream Dam Failure
- Tornado
- Earthquake
- Fire
- Loss of Fuel Pool Cooling/Makeup

From evaluation of the above events, the systems and associated system functions required to ensure safe shutdown were identified. These system functions or “modes” identified in the SSA were then systematically evaluated, and tests required to ensure capability of these functions identified. The required tests to support the SSA for each applicable system are identified in BFN Unit 1 Baseline Test Requirements Documents (BTRDs). Testing the modes identified in the SSA under the RTP at EPU conditions, as applicable, will verify the capacity of the systems to support their required safety functions.

2. Specific Acceptance Criteria

BFN UFSAR Section 13.5.2.2 presents a general description of the initial startup testing that was performed for Unit 1 and Section 13.5.2.3 for Units 2 and 3. These UFSAR sections provide the objectives and acceptance criteria for the initial startup tests. Objectives and acceptance criteria as modified to reflect operation at 120% reactor thermal power will be used for planned EPU tests.

Table 1 consolidates the initial startup tests for all three units. The table includes the tests performed at a power level of equal to or greater than 80 percent of the original licensed thermal power level as well as the tests performed at power levels lower than 80 percent. The table provides a comparison of the initial tests to the planned EPU tests; in many cases, there is a direct correlation. Initial tests which are not planned to be repeated at EPU conditions are denoted by a “No” designation in the column entitled “Testing Planned for EPU”. As denoted in the table, some of the “No” answers are for tests which are not affected by operation at EPU conditions and thus no additional justification is provided beyond that in the table. Others such as Full Main Steam Isolation Valve (MSIV) Closure and Turbine Stop/Control Valve Trips (referred to Generator Load Rejection in the ELTR) are high power tests for which the justification for not performing these tests are provided in Section III.C below.

Table 2 provides a comparison of the steady state and transient tests from the initial BFN startup tests to those described in SRP 14.2.1, Attachments 1 and 2. This table demonstrates that the applicable tests in Attachments 1 and 2 are addressed by the testing planned for BFN EPU implementation.

B. Post Modification Testing Requirements for Functions Important to Safety Impacted by EPU-Related Plant Modifications

1. General Discussion

EPU for BFN will include the implementation of several plant modifications, mostly in the balance-of-plant systems, in addition to setpoint adjustments and operating parameter changes. Individual system or component performance characteristics affected by modifications are normally demonstrated by testing required by Technical Specifications and existing 10 CFR Part 50, Appendix B, quality assurance programs. Also, routine and EPU specific planned startup testing will demonstrate system and plant acceptability.

BFN has reviewed the aggregate impact of EPU plant modifications, setpoint adjustments, and parameter changes that could adversely impact the dynamic response of the plant to anticipated initiating events. The details of this review are provided below and verify the testing program adequately demonstrates that EPU related modifications will be adequately implemented. The testing will demonstrate that functions important to safety that rely on the integrated operation of multiple systems, structures and components following an anticipated operational occurrence are capable of performing their design function prior to extended operation at the EPU power level.

2. <u>Specific Acceptance Criteria</u>

BFN has identified: a) plant modifications, b) setpoint adjustments necessary to support operation at EPU conditions and c) changes in plant operating parameters resulting from operation at EPU conditions. These modifications, setpoint adjustments and parameters have been reviewed as described below to ensure that adequate testing will be provided.

Plant modifications and setpoint adjustments necessary to support EPU for BFN Unit 1 were listed in Enclosure 7 to the original license amendment submittal. These modifications enhance, upgrade, and/or replace existing plant components to allow for operation at EPU conditions. With the exception of allowing operation at EPU conditions, these modifications do not change the design functions of the equipment or the method of performing or controlling the function. These modifications do not involve first-of-a-kind plant modifications, the introduction of new system dependencies, or changes in types of system response to initiating events.

In accordance with SRP 14.2.1, Section III.B, the EPU plant modifications, setpoint adjustments and parameter changes have been evaluated to ensure that the testing program is adequate for those that meet all of the following criteria:

a. Impacts a function important to safety

b. Is required to mitigate a plant transient listed in Attachment 2 of SRP Section 14.2.1

c. Transient/accident response involves the integrated action of multiple systems, structure and components

Table 3 tabulates the above information for the modifications, setpoint adjustments and parameter changes. The modifications have been re-ordered from that previously provided in Enclosure 7 of the original license amendment submittal to facilitate the discussion below. The modification to install safety related pressure actuation logic for the main steam relief valves has been removed from the table as it is not classified as an EPU required modification, and is not credited in the transient analyses. The modification to install five Local Power Range Monitors (LPRMs) is also removed, since this modification is not classified as an EPU required modification.

For the items evaluated in Table 3 which have a “Yes” determination for all three of the above criteria, the SRP 14.2.1, Section III.B requires an evaluation of the adequacy of the proposed EPU test program.

The items found to have a “Yes” answer for all three criteria are the feedwater/condensate pumps, the MSIVs and the Electro-Hydraulic Control (EHC) modification. An evaluation of the testing for these modifications is provided in Section III.C under the MSIV closure and turbine stop/control valve trip discussions.

The remaining items in Table 3 were evaluated and found to not meet the criteria of Section III.B.2. Further discussion of the testing associated with the items in Table 3 is provided below.

a. Modifications

The high pressure main turbine steam path and turbine sealing steam system will be modified in order to accept the higher steam flows generated at the higher power level. In addition, the low pressure turbine steam path is being replaced. The steam path modifications will not change the normal turbine control operation and will not modify or impact the turbine stop and control valves or turbine bypass valves. Both the design function of the turbines and the system response to event initiators will not change as a result of these modifications. The turbine steam flow path is not required for response to plant transients and therefore no specific testing associated with the turbines is required to demonstrate plant transient response. Thus, these modifications have no impact on the integrated plant response during transient conditions.

The condensate, condensate booster, and reactor feedwater pump modifications are performed to upgrade the components to provide the higher flows for EPU operating conditions. These pump modifications do not change the design function of the condensate/feedwater system nor will a new system interaction be created.

The moisture separators, selected feedwater heaters, main condenser extraction steam bellows, condensate demineralizers, and steam packing exhauster bypass are being upgraded to accept the EPU operating conditions. These modifications have no impact on the integrated plant response during transient conditions.

The drywell building steel, torus attached piping supports and main steam supports are being upgraded to accept the EPU transient conditions. These changes maintain acceptable design margins for the piping and support configurations. These components are passive structural elements of the plant and will receive applicable structural installation testing. These modifications have no impact on the integrated plant response during transient conditions.

The steam dryer will be modified, as required, in order to accept the higher steam flows generated at the higher power level and accommodate the effect of flow induced vibrations. As discussed in the February 23, 2005, TVA letter, the steam dryer modifications will be based on the finalized and accepted loading methodology. The steam dryer is a passive component with no associated transient response testing.

The main steam relief valves will be reset based on the higher reactor pressure. The ability of these valves to perform their transient response function is demonstrated by bench testing coupled with the fact that the valves are individually remote manually opened at reactor pressure during unit startup.

Many of the Generic Letter 89-10 valves will be replaced while the remaining 89-10 valves will be refurbished. The transient response of these valve is demonstrated through their system testing coupled with MOVATS testing.

The recirculation pump motors have been re-rated and evaluated to ensure that the motors will accommodate the EPU conditions. Operation of the recirculation pumps is not relied upon for transient response. Re-rating of the pump motors does not involve physical changes to the motors or pumps and thus, does not change their transient response characteristics such as inertia or coast down rate. Therefore, there is no testing associated with transient response for this modification. Recirculation system jet pumps will have jet pump sensing line clamps installed as required to reduce vibration from the Recirculation Pump vane passing frequencies at EPU conditions. The jet pumps are passive components with no associated transient response testing. These changes do not affect the system functions and have no impact on the integrated plant response during transient conditions.

Main generator modifications include a stator and field rewind, installation of a redesigned core end assembly, additional monitoring instrumentation, correcting all known industry identified issues, and upgrading the temperature monitoring equipment. This will allow Unit 1 to re-rate the generator from

1280 MVA to 1330 MVA. This has no effect on the generator's transient performance. Modifications to the isolated phase bus duct cooling system are limited to addition of a fan to provide redundant cooling capacity. This has no effect on the isolated phase bus system's transient performance or its response to any plant transient. The BFN switchyard is being upgraded to a double breaker scheme. These modifications were considered in the latest Grid Stability study for BFN. These modifications do not change the design function of the affected equipment, nor will a new system interaction be created. These changes have no impact on the integrated plant response during transient conditions.

Integrated Computer System (ICS)/Safety Parameter Display System (SPDS) parameters are being revised by both Technical Specification and Balance of Plant (BOP) instrument upgrades as required. The ICS/SPDS does not perform any control function but is display only. Therefore, this change has no impact on the integrated plant response during transient conditions.

Reactor Recirculation vibration monitoring sensors have been installed on Unit 2 Reactor Recirculation piping and data has been collected at 105% of OLTP conditions. Additional vibration monitoring sensors will be installed as applicable to collect and analyze vibration data on selected Unit 1 systems at EPU conditions. These are monitoring only sensors and have no impact on the integrated plant response during transient conditions.

Modifications to the main steam isolation valves are being performed to reduce differential pressure across the MSIVs and increase stem size. MSIV closure times presently specified in the Technical Specifications are being maintained at EPU conditions and are utilized in transient analyses. Post modification testing and applicable Technical Specification testing will ensure that the MSIVs will continue to meet the current closure times used in transient analyses.

While not required or being implemented specifically to support EPU operations, several additional modifications involving control system upgrades are noteworthy. The following control systems are being modified to include digital control components: Reactor Recirculation, Reactor Feedwater and Turbine (EHC). These control systems upgrades to digital was not a requirement to implement EPU; however, these modifications will ensure, from a control standpoint, that BFN Unit 1 behaves and responds consistent with BFN Units 2 and 3. The control system settings will be installed with values that support operation at EPU conditions.

The design changes for Unit 1 control system upgrades were based on Unit 3 design changes that are currently in operation. The Unit 3 design is the same as Unit 2. The digital hardware utilized in the Unit 1 control systems will be essentially the same as the digital hardware installed in Unit 3. Minor differences in internal components are the result of the manufacturers providing the most current model of the hardware. These minor hardware differences will not affect the operation of the control systems since the software will be based on the Unit 3 software. Factory acceptance tests are performed to verify the operation of the hardware and software. TVA applies a standard process to each digital software program which includes software verification and validation testing. Post-modification testing ensures that the controllers properly communicate (both input and output) with their interfacing components. Therefore, since the Unit 1 digital control systems hardware and software were developed based on the Unit 3 equivalent, the Unit 1 control systems will respond to normal operation and operational transients in the same manner as the Unit 3 control systems. The scaling changes required for Units 2 and 3 EPU will be implemented at the corresponding Units 2 and 3 EPU implementation outage in order to maintain consistency with Unit 1.

The Recirculation System Motor-Generator Sets are being replaced with Variable Frequency Drives (VFDs) and the recirculation system speed controller upgraded to a digital control system. These upgrades were previously made for Units 2 and 3. The new recirculation system controls will be configured similarly to Units 2 and 3 for 105% reactor thermal power except the appropriate inputs will be rescaled to reflect EPU operation. The VFDs provide the electrical motive power input to the two recirculation pump motors which controls the speed of each of the recirculation pump motors. Post modification testing of the speed controller and VFDs will verify the proper operation of the system to control the recirculation pump motor speeds. This post modification testing process has been used for the current operating configuration for Units 2 and 3 for both normal operation and event mitigation. The Recirculation System (including the VFDs) testing and tuning of the flow control system is performed each refueling startup during vessel hydro conditions and also at power conditions to analyze system response to speed demand of small and large changes. The recirculation system flow control capability has been proved by operation at current rated conditions for Units 2 and 3 to respond appropriately during normal operating conditions and large transient events such as turbine stop valve trips and turbine control valve trips. The analytical results for the limiting transients with the operating electrical characteristics of the upgraded recirculation flow control were provided in the PUSAR.

The Unit 1 Feedwater Control System and Reactor Feedpump Turbine Control System are being updated to a digital system; these upgrades were previously made for Units 2 and 3. The new Unit 1 digital systems will be configured similarly to Units 2 and 3 for 105% reactor thermal power except for the scaling changes required to implement EPU. As discussed in the section on operating experience for a turbine stop valve trip/turbine control valve trip, the control systems have been proven by operation at current rated conditions on Units 2 and 3 to respond appropriately to large transient events such as turbine stop valve trips and turbine control valve trips. Post modification testing of the control systems will verify the proper operation for EPU conditions. The post modification testing can be conducted by inserting simulated signals such as, for example, vessel level changes or flow demand changes. This process of using simulated signals versus conducting actual plant transients is an industry standard, and has been used for the current operating configuration. This method has proven to be effective in achieving appropriate system response for both normal operation and event mitigation.

These changes have no adverse impact on the integrated plant response during transient conditions since the control and trip circuitry (e.g., high vessel level, low pump suction pressure, etc.) will still function and prevent undesirable effects such as vessel overfill. The flow rates and response of the systems are modeled in the transient analyses. The EPU transient analyses have been performed utilizing the characteristics of the higher capacity system. Post modification testing will verify pump flow, head, and operating characteristics. EPU startup testing will perform testing of the condensate/feedwater control system, and calibration of the feedwater flow transmitters. This testing will verify the system parameters utilized in the transient analyses for the new pumps. System testing to verify the overall runout condition is not practical; however, planned post modification testing will confirm pump performance on an individual pump basis by a comparison of the design flow versus actual flow.

Feedwater/condensate pump performance will be confirmed on a system basis by comparison of data from startup testing to the calculated values. The higher flow rates will assist in the recovery from certain transients such as trip of a single feedwater, condensate booster or condensate pump. The higher capacity pumps would not significantly affect the large transients (MSIV Closure, Turbine Stop Valve Trip, Turbine Control Valve Trip). Larger capacity pumps would potentially reduce the intensity of reactor vessel level decrease transients but generally have little effect in the short deviations in available flow and reactor water level.

The Unit 1 EHC system is being updated to a digital system; this upgrade was previously made for Units 2 and 3. The new Unit 1 digital system will be configured similarly to Units 2 and 3 for 105% reactor thermal power except for the scaling changes required to implement EPU. The scaling changes required for Units 2 and 3 EPU will be implemented at the corresponding Units 2 and 3 EPU implementation outage, which will ensure consistency with Unit 1. The EHC system controls the turbine and auxiliaries, including the turbine stop and control valves. During plant events requiring isolation of the steam lines to the main turbine, the EHC system provides the input to close the turbine stop valves. As discussed in the section on operating experience for a turbine stop valve trip/turbine control valve trip, the EHC software has been proven by operation at current rated conditions on Units 2 and 3 to respond appropriately to large transient events such as turbine stop valve trips and turbine control valve trips. Post modification testing of the EHC modifications will verify the proper operation of the EHC system, associated turbine controls and proper communication with the Reactor Protection System (RPS) for EPU conditions. The post modification testing can be conducted by inserting simulated signals such as low EHC pressure and stop valve position. This process of using simulated signals versus conducting actual plant transients has been used for the current operating configuration and has proven to be effective in achieving appropriate system response for both normal operation and event mitigation. The EHC modifications will have insignificant impact on transient system response.

The original Unit 1 Reactor Protection System (RPS) actuation instrumentation and Unit 1 Emergency Core Cooling System (ECCS) actuation instrumentation were replaced with analog trip systems as was previously performed for Units 2 and 3. The Units 2 and 3 analog trip systems design has been in operation for several years and has continued to provide reliable inputs to the actuation logic. The Unit 1 design provides physical and electrical separation as required to ensure high reliability and diversity while maintaining the original system actuation logic. Post modification testing validates the installation of the new instrument designs and ensures the systems will perform as required.

Based on the Unit 1 digital controls system software/hardware, and RPS and ECCS analog trip system design similarity with Units 2 and 3, the response of the Unit 1 instrumentation and controls to normal operation and transients should reflect the instrumentation and controls of Unit 3 response and thus the operational experience from Units 2 and 3 demonstrates that Unit 1 will respond properly to expected transients.

b. Setpoint Adjustments

Technical Specification instruments have been reviewed for range/span and applicability at EPU conditions. Design changes have been prepared which will replace/re-span instruments as required to ensure proper instrument loop performance. The design modification process ensures appropriate testing for each loop when modified. The revised limits and setpoints are included in the transient analyses and have been evaluated as acceptable for EPU operation. Plant control systems for Feedwater, Recirculation, and the turbine will be upgraded to digital control systems as described in Section III.B.2.a above. The new systems will contain the required input for implementation of EPU so that the systems will be functionally identical to Units 2 and 3 after EPU implementation. Therefore, these changes will ensure proper transient system response.

Balance of plant (BOP) instruments have been reviewed for range/span and applicability at EPU conditions. Design changes have been prepared which will replace/re-span instruments as required to ensure proper instrument loop performance. The design modification process ensures appropriate testing for each loop when required. This change has no impact on the integrated plant response during transient conditions.

c. Changes in Plant Operating Parameters

For BFN EPU, the increase in electrical output is accomplished primarily by generation and supply of higher steam flow for the turbine generator. The high steam production is achieved by increasing the reactor power along slightly modified rod and core flow control lines. However, there is no increase in the maximum allowable recirculation flow value.

PUSAR Table 1-2 provides a summary of the reactor thermal-hydraulic parameters for the current rated and EPU conditions. Thermal power, reactor pressure, main steam flow, feedwater flow, and feedwater temperature are the only operating parameters with a significant change that have a potential impact on transient analyses.

As seen in the discussion of planned modifications above, many of the planned modifications are related to the change in the reactor pressure, main steam and feedwater operating parameters. Accordingly, these operating parameter changes are being appropriately incorporated into plant systems and are being evaluated and tested as part of those changes as described in Table 3. Additionally, the transient and accident analyses performed for EPU were evaluated utilizing the changed operating parameters for EPU conditions. The results of the limiting transients were provided in the PUSAR. Parameters such as post accident containment pressure and torus temperature are not explicitly tested but are instead accounted for analytically in establishing the testing requirements.

Therefore, the changes in plant operating parameters resulting from operation at EPU conditions have been appropriately identified, evaluated, and incorporated in planned post modification testing and EPU startup testing.

C. Use of Evaluation to Justify Elimination of Power-Ascension Tests

1. General Discussion

As discussed above in Sections III.A.1 and III.B.1, Tables 1, 2 and 3 define the testing that was accomplished for the respective unit startup and that planned for EPU related modifications. The tables also compare the BFN EPU testing to the testing proposed by SRP 14.2.1. In some cases, there are differences or exceptions in the BFN testing compared to the SRP proposed testing. The following issues require further evaluation and justification for not performing these tests in accordance with SRP Section III.C:

STP 25 Main Steam Isolation Valves
STP 27 Turbine Stop and Control Valve Trips

2. Specific Acceptance Criteria

The following provides an evaluation for each of the above listed tests against the specific acceptance criteria in SRP 14.2.1, Section III.C.2.

STP 25 Main Steam Isolation Valves

This initial startup test required a simultaneous full closure of all MSIV's at about 100 percent of rated thermal power to demonstrate proper operation of the relief valves and the Reactor Core Isolation Cooling (RCIC). Since it is not intended to perform a similar test at EPU implementation, the following provides the evaluation of the acceptance criteria provided in SRP Section III.C.

The MSIV closure is an Anticipated Operational Transient as described in Chapter 14 of the BFN UFSAR. The MSIV closure is classified as an event that results in a sudden reduction of steam flow while the reactor is operating at power and, therefore, a significant nuclear system pressure increase. The closure of all MSIVs with direct scram failure (reactor scram on high neutron flux signal rather than the MSIV position switches) is the design basis event to analytically demonstrate compliance with the American Society of Mechanical Engineers (ASME) vessel overpressure protection criteria (upset condition). The test proposed in the UFSAR by the initial startup test is the complete closure of all MSIVs with direct scram (scram on MSIV position switches).

The original MSIV closure tests for full isolation (closure of all eight valves simultaneously) are described in UFSAR Sections 13.5.2.2 as Test Number 25, and were intended to demonstrate the following:

1. The transient rise in simulated heat flux shall not exceed 10%,
2. The initial transient rise in vessel dome pressure occurring within 20 seconds of the main steam isolation valve trip initiation shall not exceed 150 psi,
3. MSIV closure time must be greater than 3 and less than 5 seconds,
4. Correct performance of the RCIC System,
5. Correct performance of the Main Steam Relief Valves.

For Item 1, the intent was to monitor fuel performance. For this event, the closure of the MSIVs causes a vessel pressure increase and an associated increase in reactivity. The negative reactivity of the Reactor Protection System (RPS) scram from MSIV position switches offsets the positive reactivity of the pressure increase such that there is a minimal increase in heat flux. EPU will have minimal impact on the components important to achieving the desired thermal performance. RPS logic is unaffected and as discussed in PUSAR Section 2.5.1, overall control rod insertion times will not be significantly affected. MSIV closure speed is controlled by adjustments to the actuator and can be adjusted to occur within the required closure times.

The MSIV limit switches that provide the scram signal to the RPS are highly reliable devices that are suitable for all aspects of this application including environmental qualification requirements. There is no direct effect by any EPU changes on these switches or the RPS. Similarly there is no impact on the instrumentation actuated by reactor vessel pressure (other than rescaling). There may be an indirect impact caused by slightly higher ambient temperatures, but the increased temperatures will still be below the qualification temperature. These instruments are expected to be equally reliable before and after EPU. The RPS logic is unaffected by EPU, the MSIV closure time will not be affected by EPU, and the switches which initiate the scram and Recirculation Pump trip are highly reliable.

The increased post event pressure associated with the MSIV closure would not adversely impact the recirculation pump coastdown characteristics since the pressure acts on both sides of the pump. Consequently, the impact on transient heat flux results due to EPU is acceptable.

For Items 2 and 5, due to the minimal nature of the flux transient, the expected reactor pressure rise is largely dependent on Main Steam Relief Valve (MSRV) setpoints and initial reactor pressure. As discussed in the PUSAR, Section 3.1, the MSRV setpoints are increased consistent with the operating dome pressure increase. The uprated nominal MSRV setpoints ensure that adequate differences (simmer margin) between operating pressure and MSRV setpoints are maintained. Therefore, there is no effect on valve functionality (opening/closing).

As discussed below, in the section on operating experience for turbine stop valve trip/turbine control valve trip, the existing Units 2 and 3 MSRVs performance has proven acceptable during BFN transients. The most recent examples include:

- The Unit 2 May 15, 1999, Turbine Trip from 105% OLTP (LER 260/1999-003-01) where reactor pressure increased due to stop valve closure and resulted in five MSRVs opening initially. Pressure was then controlled by the turbine bypass system, and all systems responded as expected.

- The Unit 2 July 25, 2001, Turbine Control Valve Fast Closure Scram from 105% OLTP (LER 260/2001-003-00) where eight MSRVs opened initially until pressure was controlled by the turbine bypass system, and all systems responded as expected.

- The Unit 2 July 27, 2002, Generator Load Rejection from 105% OLTP (LER 260/2002-002-00) where four MSRVs opened initially until pressure was controlled by the turbine bypass system, and all systems responded as expected.

- The July 8, 2004, Unit 2 main turbine trip from 105% OLTP (LER 260/2004-001-00) where seven MSRVs opened initially until pressure was controlled by the turbine bypass system, and all systems responded as expected.

The MSRVs are tested each refueling outage in accordance with the Technical Specifications setpoint verification and through remote manual opening at reactor operating pressure. Performance of an actual MSIV closure test would provide no benefit for demonstrating the capability of the MSRVs for vessel overpressure protection that is not already accomplished by this component level testing.

For Item 3, the focus of the original testing was to verify that the steam flow from the reactor was not shut off faster than assumed (i.e., 3 seconds), since steam flow assists MSIV closure. During maintenance and surveillance, MSIV actuators are adjusted as necessary to control closure speed, and BFN test performance has been within the acceptance criteria. As discussed in Section III.B of this enclosure, the MSIVs are being modified to reduce the pressure drop across the valves. Procedurally required testing will ensure the valves continue to meet the Technical Specification required closure limits. The BFN MSIVs were evaluated for EPU and determined acceptable for EPU operation. Industry experience, including BFN, has shown that there are no significant generic problems with this design.

For Item 4, a MSIV closure test is not required to demonstrate acceptable RCIC performance. During an MSIV closure event, the RCIC System would not initiate until the reactor vessel level decreased to the low low level (level 2) setpoint. This would not occur immediately since the level would decrease slowly due to relief valve actuations alone. This is not a rapid event in terms of RCIC response, the event does not produce any unique operating situations that have not been demonstrated previously by operation of the RCIC System, and there is no functional dependency between the closure of the MSIVs and the successful operation of the RCIC System. Thus, an MSIV closure test would not challenge the actuation logic beyond that exercised during periodic Technical Specification testing. Since there is a change to the normal reactor operating pressure and the MSRV setpoints change, the RCIC System performance parameters and associated setpoints will be rescaled accordingly as discussed in PUSAR Section 3.8. Operating history on Units 2 and 3 demonstrate that rescaling the system based on the increased pressure maintains full system functional capability over the expected pressure range. The rescaled system parameters will limit the maximum peak RCIC turbine speed. Additional confidence in this rescaling approach is provided by the operating experience on Units 2 and 3 for the rescaling on those units. As part of the power ascension testing for the 105% uprate on both Units 2 and 3, a RCIC simulated automatic cold quick start was performed when the reactor was within the uprated operating pressure window as defined by Technical Specifications. Since that time, RCIC has been routinely tested to assure that it can deliver rated flow and pressure and has initiated automatically during an operating event at the low low reactor vessel level as discussed below in the turbine stop valve trip/turbine control valve trip operating experience section. These items are adequate to show that RCIC can reliably deliver rated flow and perform its automatic function.

The following addresses the seven factors outlined in SRP 14.2.1, Section III.C.2.

a. Previous Operating Experience

BFN Unplanned Power Uprate Related Transients

Since implementation of the original 105% power uprate, BFN Units 2 and 3 have not experienced an MSIV closure event from near full power. However, as discussed for the Turbine Stop and Control Valve Trip test (STP 27), several events have occurred which have successfully demonstrated operation of the RPS, control rod drives, MSRVs and RCIC.

Industry Unplanned Power Uprate Related Transients

To date, thirteen other plants have implemented EPUs:

- Hatch Units 1 and 2 (from 105% to 113% of OLTP)
- Monticello (from 100% to 106.3% OLTP)
- Muehleberg (i.e., KKM) (from 105% to 116% OLTP)
- Leibstadt (i.e., KKL) (from 105% to 117% OLTP)
- Duane Arnold (from 105% to 120% OLTP)
- Brunswick Units 1 and 2 (from 105% to 120% OLTP)
- Quad Cities Units 1 and 2 (from 100% to 117% OLTP)
- Dresden Units 2 and 3 (from 100% to 117% OLTP)
- Clinton (from 100% to 120% OLTP)

The following event involving an MSIV closure was experienced at the Hatch 2 plant. Hatch 2 experienced a reactor trip on high reactor pressure as a result of MSIV closure (from 113% OLTP (100% of uprated power)) in 2001. As noted in Hatch 2 LER 2001-003-00, systems functioned as expected and designed, given the conditions experienced during the event.

b. <u>Introduction of New Thermal-Hydraulic Phenomena or Identified System Interactions</u>

As discussed earlier, no modifications are to be performed as part of EPU implementation that would cause BFN to behave differently from previous operating experience for an MSIV closure event. Related to the MSIV closure, EPU will have no impact on the components important to achieving the desired thermal performance. RPS logic is unaffected and as discussed in PUSAR Section 2.5.1, overall control rod insertion times will not be significantly affected. The event does not produce any unique operating situations that have not been demonstrated previously by operation of the RCIC system and there is no functional dependency between the closure of the MSIVs and the successful operation of the RCIC system. As discussed above, the MSIVs were evaluated for EPU and are acceptable for EPU operation. MSIV closure speed is controlled by adjustments to the actuator and is considered very reliable. As discussed in the PUSAR, Section 3.1 the MSRV setpoints are increased consistent with the operating dome pressure increase. The uprated nominal MSRV setpoints ensure that adequate differences (simmer margin) between operating pressure and MSRV setpoints are maintained. There is no effect on valve functionality (opening/closing). Therefore, there are no new thermal-hydraulic or identified system interactions.

c. <u>Facility Conformance to Limitations Associated With Analytical Analysis Methods</u>

The safety analyses performed for BFN used NRC-approved transient modeling codes. The NRC has accepted these for BWRs with a range of power levels and power densities that bound the requested power uprate for BFN. The codes have been benchmarked against BWR test data and have incorporated industry experience gained from previous transient modeling. Analyses use plant specific inputs and models all the essential physical phenomena for predicting integrated plant response to the analyzed transient. Thus, the codes will accurately and/or conservatively predict the integrated plant response to this transient at EPU power levels and no new information about transient modeling is expected to be gained from performing this large transient test.

d. <u>Plant Staff Familiarization With Facility Operation and Trial Use of Operating and Emergency Operating Procedures</u>

As discussed in Section 10.6 of the PUSAR, some additional training is required to enable plant operation at EPU conditions and power level. For EPU conditions, the scope of operator responses to transient, accident and special events are not affected.

e. <u>Margin Reduction in Safety Analysis Results for Anticipated Operational Occurrences</u>

The limiting pressurization transient events at EPU are the MSIV closure and turbine trip with turbine bypass failure. Both events are analyzed with failure of direct scram (i.e., scram based on neutron flux instead of valve position). BFN MSIV closure analyses assume that the events initiate at 102% of EPU reactor thermal power, the reactor dome pressure is 1055 psig (which is 20 psi higher than the nominal EPU dome pressure), one MSRV (with the lowest setpoint) is out-of-service (OOS) and uses a standard MSIV closure profile (percent area closure versus time) which has been shown by analysis results for the 105% power level to increase the predicted peak pressure by approximately 14 psi compared to the BFN specific closure profile. The 20 psi increase in initial starting pressure coupled with the 14 psi from the closure profile results in the peak predicted pressure being conservatively increased approximately 34 psi between 105% and 120% power.

The results of the EPU overpressure protection analysis are given in Figure 3-1 of the PUSAR. The peak reactor vessel pressure is the only parameter for this transient that has a specific NRC acceptance limit. The calculated peak reactor pressure vessel (RPV) pressure remains below the ASME limit, and the maximum calculated dome pressure remains below the Technical Specifications Safety Limit. However, the pressure increase does exceed the NRC definition of a minimum change in consequences of no more than a 10% reduction in margin.

Even though the "less than 10% reduction in margin" criterion would be exceeded, the performance of the actual test would not yield a substantial amount of confirmatory data. The actual test would be conducted in the following manner:

- Using a direct scram based on the MSIV position switches which reduces the resulting pressure rise by approximately 100 psi,

- Using the BFN specific MSIV closure profile which would reduce the reactor vessel pressure rise by 14 psi,

- Using an initial reactor dome pressure of 1035 psig which would provide a 20 psi margin increase,

- All SRVs would be in service which would reduce the peak reactor vessel pressure by approximately 10 psi,

- The reactor thermal power would be limited to 100% of rated.

As discussed above, the performance of the RCIC system is not substantially challenged by this test. The system is tested by existing plant procedures that duplicate expected operational conditions during plant transients to demonstrate acceptable initiation and operation. No new information with regard to transient modeling or the analysis results are expected to be gained from performing this large transient test.

Therefore, TVA concludes that performance of a full MSIV closure transient test would not yield sufficient information to warrant conducting this test.

f. <u>Guidance Contained in Vendor Topical Reports</u>

The EPU license application was prepared following the guidelines contained in the NRC approved General Electric (GE) Company Licensing Topical Reports NEDC 32424P-A (ELTR1), February 1999, and NEDC 32523P-A (ELTR2), February 2000, and its Supplement 1, Volumes I and II. Appendix L, Section L.2.4 of ELTR1 currently discusses the need to perform large transient testing, specifically a Main Steam Isolation Valve (MSIV) closure test and a Generator Load Rejection test. Therefore, as indicated for these tests in Table 1, a BFN plant-specific basis for exception to re-performing the original startup large transient tests is provided.

g. <u>Risk Implications</u>

While this application is not "risk-informed," TVA believes, on a qualitative basis, that the benefits from performing these tests are negligible, when assessed against the risks of subjecting the plant to an otherwise unnecessary challenge. A scram from high power level results in an unnecessary and undesirable transient cycle on the primary system. In addition, the risk posed by intentionally initiating a large transient event, although small, should not be incurred unnecessarily.

<u>STP 27 Turbine Stop and Control Valve Trips</u>

During performance of STP 27, the turbine stop and control valves were tripped at selected reactor power levels (50% and 100%). Several reactor and turbine operating parameters were monitored to evaluate the response of the bypass valves, relief valves, and the RPS. The peak values and change rates of reactor steam pressure and heat flux were determined.

Since TVA does not intend to perform a similar test at EPU implementation, the following provides the evaluation of the acceptance criteria provided in SRP Section III.C.

The Turbine Stop Valve Trips and Turbine Control Valve Trip events are classified as an Anticipated Operational Transient as described in Chapter 14 of the BFN UFSAR. The events are classified as an event that results in a sudden reduction of steam flow while the reactor is operating at power and, therefore, a significant nuclear system pressure increase.

A turbine stop valve trip from high power conditions is the result of a turbine or reactor system malfunction which produces the following transient sequence:

a. Turbine Stop Valves fast closure (0.1 second closure time) which produces a fast steam flow shutoff,

b. Position switches on the stop valves sense the trip and initiate immediate reactor scram (for initial power levels above 30 percent),

c. The turbine bypass valves are opened simultaneously with turbine control valve closure, and reroute a portion of vessel steam flow to the condenser,

d. Reactor vessel pressure rises to the MSRV setpoints, causing them to open for a short period of time,

e. The steam passed by the MSRVs is discharged into the suppression pool, and

f. The turbine bypass valve (TBV) system controls nuclear system pressure after the MSRVs close.

A turbine control valve trip from high power conditions produces the following transient sequence:

a. Turbine-generator power/load unbalance circuitry and other generator trips initiate turbine control valve (TCV) fast closure (minimum response time of TCV fast closure: 0.15 seconds),

b. Turbine control valve fast closure is sensed by the reactor protection system, which initiates a scram and simultaneous recirculation pump trip (for initial power levels above 30 percent rated),

c. The turbine bypass valves are opened simultaneously with turbine control valve closure, and reroute a portion of the vessel steam flow to the condenser,

d. Reactor vessel pressure rises to the MSRV setpoints, causing them to open for a short period of time,

e. The steam passed by the MSRVs is discharged into the suppression pool, and

f. The turbine bypass valve (TBV) system controls nuclear system pressure after the MSRVs close.

The original turbine stop/control valve closure tests are described in UFSAR Section 13.5.2.2 as Test Number 27, and were intended to demonstrate the following:

1. The transient rise in simulated heat flux shall not exceed 10 percent.

2. The initial transient rise in vessel dome pressure occurring within 10 seconds of the turbine/generator trip initiation shall not be greater than 150 psi.

3. Correct performance of the main steam relief valves.

4. Correct performance of the turbine bypass valves.

5. The turbine stop valves must begin to close before the control valves for the turbine trip. The turbine control valves must begin to close before the stop valves during the turbine control valve trip.

6. Following fast closure of the turbine stop and control valves, a reactor scram shall occur if the turbine first stage pressure is greater than the pressure corresponding to 30% power (26% following EPU implementation).

7. Feedwater systems must prevent flooding of the steamline following the transients.

8. The pressure regulator must prevent a low-pressure reactor isolation. The feedwater controller must prevent a low-level initiation of the HPCI System and MSIV isolation as long as feedwater remains available.

For Item 1, the intent was to monitor fuel performance. For these events, the closure of the stop/control valves causes a vessel pressure increase and an increase in reactivity. The negative reactivity of the scram would offset the positive reactivity of the pressure increase such that there is a minimal increase in heat flux. EPU will have minimal impact on the components important to achieving the desired thermal performance.

The EHC system is being updated to a digital system; this upgrade was previously made for Units 2 and 3. The new Unit 1 digital system will be configured similarly to Units 2 and 3 for 105% reactor thermal power except for the scaling changes required to implement EPU. The scaling changes required for Units 2 and 3 EPU will be implemented at the corresponding Units 2 and 3 EPU implementation outage, which will ensure consistency with Unit 1. As discussed in the section on operating experience for a turbine stop valve trip/turbine control valve trip, the EHC software has been proven by operation at current rated conditions on Units 2 and 3 to respond appropriately to large transient events such as turbine stop valve trips and turbine control valve trips. Post modification testing of the EHC modifications will verify the proper operation of the EHC system, associated turbine controls and proper communication with the RPS for EPU conditions. The post modification testing can be conducted by inserting simulated signals such as low EHC pressure and stop valve position. This process of using simulated signals versus conducting actual plant transients has been used for the current operating configuration and has proven to be effective in achieving appropriate system response for both normal operation and event mitigation. The EHC modifications will have insignificant impact on transient system response.

RPS logic is unaffected and as discussed in PUSAR Section 2.5.1, overall control rod insertion times will not be significantly affected. The switches that provide the scram signal are highly reliable devices that are suitable for all aspects of this application. There is no direct effect by any EPU changes on these switches. There may be an indirect impact caused by slightly higher ambient temperatures, but the increased temperatures will still be below the functional temperature range of the components. These switches are expected to be equally reliable before and after EPU.

The increased post event pressure associated with the turbine stop valve trip/turbine control valve trip would not adversely impact the recirculation pump coastdown characteristics since the pressure acts on both sides of the pump. Consequently, there is no effect on fuel thermal performance or heat flux results due to EPU.

For Items 2, 3 and 4, due to the minimal nature of the flux transient, the expected reactor pressure rise is largely dependent on MSRV and turbine bypass valve performance. As discussed in the PUSAR, Section 3.1 the MSRV setpoints are increased consistent with the operating dome pressure increase. The uprated nominal MSRV setpoints ensure that adequate differences (simmer margin) between operating pressure and MSRV setpoints are maintained. Therefore, there is no effect on valve functionality (opening/closing). The turbine bypass valves will be controlled by the digital EHC system with settings appropriate for EPU conditions.

As discussed in the below section on operating experience, the existing MSRVs performance as well as the performance of the turbine bypass valves has proven acceptable during BFN transients. The most recent examples include:

- The Unit 2 May 15, 1999 Turbine Trip from 105% OLTP (LER 260/1999-003-01) where reactor pressure increased due to stop valve closure and resulted in five MSRVs opening initially. Pressure was then controlled by the turbine bypass system, and all systems responded as expected.

- The Unit 2 July 25, 2001 Turbine Control Valve Fast Closure Scram from 105% OLTP (LER 260/2001-003-00) where eight MSRVs opened initially until pressure was controlled by the turbine bypass system, and all systems responded as expected.

- The Unit 2 July 27, 2002 Generator Load Rejection from 105% OLTP (LER 260/2002-002-00) where four MSRVs opened initially until pressure was controlled by the turbine bypass system, and all systems responded as expected.

- The July 8, 2004 Unit 2 main turbine trip from 105% OLTP (LER 260/2004-001-00) where seven MSRVs opened initially until pressure was controlled by the turbine bypass system, and all systems responded as expected.

The MSRVs are tested each refueling outage in accordance with the Technical Specifications setpoint verification and through remote manual opening at operating reactor pressure. The turbine bypass valves are exercised during startup to control reactor pressure which confirms their capability to perform under reactor operating pressure conditions. Performance of an actual stop/control valve closure test would provide no benefit for demonstrating the capability of the MSRVs or turbine bypass valves for vessel overpressure protection that is not already accomplished by this component level testing.

For Item 5, during normal operation, the EHC emergency trip fluid system (ETS) is pressurized to open and control the main turbine steam valves. When a turbine trip occurs, the ETS is depressurized removing EHC fluid from the turbine stop valves and relay trip valve. The relay trip valve then removes EHC fluid from the control valve allowing it to close. These physical attributes of the EHC system have not been modified and are not affected by pressure or flow changes associated with EPU.

The signal that initiates a turbine control valve trip is a power load imbalance. This signal is used to anticipate a rapid acceleration of the turbine before a measurable increase in speed is detected. When this occurs, the control valve fast acting solenoids open removing EHC fluid from the control valves only. The control valves close to prevent turbine acceleration. This control function has not been modified since the original test and is not affected by pressure or flow changes associated with EPU.

For item 6, the intent was to demonstrate that a scram would occur based on stop/control valve position if the reactor power is above the low power setpoint. As demonstrated in the below section on Units 2 and 3 operating experience, a scram occurred in each event which confirms the functionality of the scram initiation on valve position. The conditions associated with EPU do not affect the capability of the electronic sensing system.

For items 7 and 8, the intent was to demonstrate that the feedwater control system and the pressure regulator were capable of controlling the post event pressure and reactor vessel water level such that no reactor vessel flooding, no low water level initiations of HPCI and no reactor vessel isolation occurred. As demonstrated in the below section on operating experience, BFN Units 2 and 3 have experienced several events of this type with successful operation of the feedwater controller and pressure regulator. The controller adjustments to account for EPU conditions have been included in the transient analyses and demonstrated to provide proper control such that the acceptance criteria will be met. The modifications will have post-modification testing to confirm the proper operational characteristics. The conditions associated with EPU do not affect the capability of the systems to perform these functions.

The following addresses the seven factors outlined in SRP 14.2.1, Section III.C.2.

a. <u>Previous Operating Experience</u>

<u>BFN Unplanned Power Uprate Related Transients</u>

Since implementation of the original 105% power uprate, BFN Units 2 and 3 have previously experienced the following unplanned high power level large scale operating transients (100% power/automatic scram events):

- On May 15, 1999, Unit 2 received an automatic scram from 100 percent reactor power due to a turbine trip that occurred during routine turbine overspeed testing. The reactor scram caused reactor water level to go below the low level setpoint (level 3) which generated a redundant scram signal and initiated the PCIS as expected. The low reactor water level signal also initiated the SGTS and CREVS. The transient was mitigated by the automatic opening of nine Main Steam Bypass Valves and five Main Steam Relief Valves. All systems responded as expected and all control rods fully inserted. The cause of the turbine trip was failure of the mechanical trip cylinder to latch when hydraulically reset. This event was reported to the NRC in LER 1999-003-00, dated 6/14/99.

- On May 24, 2000, with Unit 3 operating at 100 percent power, an invalid low reactor water level scram signal was generated while returning a feedwater level transmitter to service following scheduled calibration. The reactor scram caused reactor water level to decrease below the low level (level 3) and low-low level (level 2) setpoints. All emergency systems operated as expected in response to the scram, including initiation of HPCI and RCIC and insertion of all control rods. This event did not result in the loss of the normal heat removal path since the condenser remained available throughout the event and was used for decay heat removal and no Main Steam Relief Valves opened. The scram was the result of a pressure perturbation in the common variable sensing line shared with both channels of reactor protection system level transmitters. This event was reported to the NRC in LER 2000-001-00 dated 6/22/00.

- On July 25, 2001, Unit 2 received an automatic scram from 100 percent power due to a main turbine trip from a power-load unbalance that occurred during Combined Intermediate Valve Testing. The vendor software contained a numerical error that resulted in an inadvertent turbine trip. The reactor scram caused the water level to go below the low level setpoint (level 3) which generated an additional scram signal and initiated PCIS. Following the initial pressure transient which peaked at 1148 psig, eight Main Steam Relief Valves opened. All systems responded as expected and all control rods fully inserted. This event was reported to the NRC in LER 2001-003-00 dated 9/21/01.

- On July 27, 2002, a Unit 2 main generator trip, main turbine trip, and reactor scram occurred from 100% power. All expected system responses were received, including the automatic opening of four safety relief valves. Actuation of PCIS groups 2, 3, 6, and 8 occurred due to the expected temporary lowering of reactor water level. The normal heat rejection path for the reactor remained in service. Reactor water level was recovered to the normal operating range by the normal reactor water level control system. All systems responded as expected and all control rods fully inserted. This event was reported to the NRC in LER 2002-002-00.

- On July 8, 2004, a Unit 2 main turbine trip/reactor scram occurred from 100% power. All expected system responses were received, including the automatic opening of seven safety relief valves. Electrical switching was in progress at the time of the scram and during this switching activity, the Unit 2 UPS 120VAC Bus was inadvertently de-energized briefly. The reactor scram occurred at this time due to a turbine control valve fast closure/turbine trip condition. The loss of the UPS power would not itself be expected to result in a turbine trip/reactor scram because of the fault-tolerant design of the main turbine EHC system logic. However, it was determined that one of two main generator output current signal channels in the EHC logic has been automatically bypassed previously by the system software during a separate power supply transient on a different plant distribution bus. The subsequent temporary interruption of the UPS bus caused the loss of the second main generator output current signal channel and the system logic indicated that a power-load unbalance (i.e., main generator load reject) condition existed. This event was reported to the NRC in LER 2004-001-00.

- On November 23, 2004, while Unit 3 was in steady state operation at 100 percent power, a main turbine trip and subsequent reactor scram occurred. All expected system responses occurred. A lightning strike occurred on the TVA 500-kV system approximately 40 miles distant from Browns Ferry. This strike resulted in a phase-to-ground fault on all three phases of the transmission line and the electrical power transient caused speed perturbations on both the Unit 2 and Unit 3 main turbines. The rate of speed change seen on Unit 3 was slightly greater than the maximum rate anticipated by the turbine control system logic and therefore, the turbine speed feedback signals, while valid, were designated as invalid by the logic. With all turbine speed feedback signals designated as invalid, a main turbine trip on loss of speed feedback occurred in accordance with system design and a reactor scram occurred due to the turbine trip. All expected system responses occurred. No safety relief valve operation occurred during the trip transient and post-trip review confirmed that peak reactor pressures remained below the nominal SRV lift setpoints. This event was reported to the NRC in LER 2004-002-00.

- On February 11, 2005, the Unit 3 reactor scrammed from 100% power. The scram was caused by a simultaneous false trip signal generated to the main generator circuit breaker 234, switchyard circuit breakers 5264 and 5268, and a main generator trip. This signal was generated when a PK block (disconnect device 26W), which had been pulled as part of a clearance for breaker 5264, was re-inserted as part of a switching order from the Load Dispatcher for returning the breaker to service. When the PK block 26W was inserted (out of sequence of the switching order), the associated current transformer circuit was momentarily grounded resulting in a false differential. The correct sequence of the switching order was to actuate the trip cutout switches for the differential trip functions prior to inserting any of the PK blocks. The generator trip resulted in a turbine trip and opening of the output breakers caused a power-load unbalance trip. The control valve fast closure caused the reactor to scram. All rods inserted. Reactor water level lowered, as expected, and was recovered by normal feed water flow. All expected Primary Containment Isolation System isolations were received along with the auto start of Control Room Emergency Ventilation, and the three Standby Gas Treatment trains. This event was reported in LER 2005-001-00.

As reflected in the events discussed above, BFN has experienced unplanned pressurization transients at approximately 3458 megawatts thermal (MWt). No abnormalities or deviations from predicted behavior were observed and no significant anomalies were seen in BFN's response to these large scale transient events. Since the BFN Unit 1 uprate included the same reactor vessel steam dome pressure changes as previously implemented for Units 2 and 3, this experience is applicable to EPU.

Industry Unplanned Power Uprate Related Transients

To date, thirteen other plants have implemented EPUs:

- Hatch Units 1 and 2 (from 105% to 113% of OLTP)
- Monticello (from 100% to 106.3% OLTP)
- Muehleberg (i.e., KKM) (from 105% to 116% OLTP)
- Leibstadt (i.e., KKL) (from 105% to 117% OLTP)
- Duane Arnold (from 105% to 120% OLTP)
- Brunswick Units 1 and 2 (from 105% to 120% OLTP)
- Quad Cities Units 1 and 2 (from 100% to 117% OLTP)
- Dresden Units 2 and 3 (from 100% to 117% OLTP)
- Clinton (from 100% to 120% OLTP)

Southern Nuclear Operating Company's (SNOC) application for EPU of Hatch Units 1 and 2 was granted. BFN and Hatch are both BWR/4 with Mark 1 containments. Hatch Unit 2 experienced an unplanned event that resulted in a generator load reject from approximately 111% OLTP (98% of uprated power) in the summer of 1999. As noted in SNOC's LER 1999-005-00, no anomalies were seen in the plant's response to this event. In addition, Hatch Unit 1 has experienced two turbine trips from 112.6% and 113% of OLTP (99.7% and 100% of uprated power) as reported in LERs 2000-004-00 and 2001-002-00, respectively. Again, the behavior of the primary safety systems was as expected. No new plant behaviors for either plant were observed. This indicates that the analytical models being used are capable of modeling plant behavior at EPU conditions.

The KKL power uprate implementation program was performed during the period from 1995 to 2000. Power was raised in steps from its previous operating power level of 3138 MWt (i.e., 104.2% of OLTP) to 3515 MWt (i.e., 116.7% OLTP). Uprate testing was performed at 3327 MWt (i.e., 110.5% OLTP) in 1998, 3420 MWt (i.e., 113.5% OLTP) in 1999 and 3515 MWt in 2000.

KKL testing for major transients involved turbine trips at 110.5% OLTP and 113.5% OLTP and a generator load rejection test at 104.2% OLTP. The KKL turbine and generator trip testing demonstrated the performance of equipment that was modified in preparation for the higher power levels. Equipment that was not modified performed as before. The reactor vessel pressure was controlled at the same operating point for all of the uprated power conditions. No unexpected performance was observed except in the fine-tuning of the turbine bypass opening that was done as the series of tests progressed. These large transient tests at KKL demonstrated the response of the equipment and the reactor response. The close correlation to the predicted response provides additional confidence that the uprate licensing analyses consistently reflected the behavior of the plant.

Progress Energy's Brunswick Units 1 and 2 were licensed to 120% of OLTP. BFN and Brunswick are BWR/4 plants with Mark I containments. Brunswick Unit 2 experienced an unplanned event that resulted in a generator/turbine trip due to loss of generator excitation from 115.2% OLTP (96% of uprated thermal power) in the fall of 2003. As noted in Progress Energy's LER 2003-004-00, no anomalies were experienced in the plant's response to this event. No new plant behaviors were observed. This indicates that the analytical models being used are capable of modeling plant behavior at EPU conditions.

BFN and the Exelon Generation Company's Quad Cities Units 1 and 2 and Dresden Units 2 and 3 units are BWR/3 or BWR/4 plants with Mark 1 containments. Dresden 3 has experienced several turbine trips and a generator load rejection from high uprated power conditions. In January of 2004, Dresden 3 experienced two turbine trips from 112.3% and 113.5% of OLTP (96% and 97% of uprated power) as reported in LERs 2004-001-00 and 2004-002-00, respectively. The plant response was as expected and no new plant behaviors were observed. This indicates that the analytical models used for transient analyses are capable of modeling plant behavior at EPU conditions. In May 2004, Dresden 3 also experienced a loss of offsite power which resulted in a turbine trip on Generator Load Rejection from 117% of OLTP (100% of uprated power). Control rods fully inserted, and system and containment isolations occurred as expected. Manual initiations proceeded in accordance with procedures. Plant response indicates that the analytical models being used are capable of modeling plant behavior at EPU conditions; however, there were several failures (unrelated to the analysis models) involving the Standby Gas Treatment System and an Emergency Diesel Generator output breaker. This was reported in LER 2004-003-00.

Data collected from testing and responses to unplanned transients for Hatch Units 1 and 2, Brunswick 2, Dresden 2 and 3, and KKL plants during post-EPU operation have shown that plant response has consistently been as planned, within expected parameters, and bounded by the plant transient and safety analyses. Based on the similarity in design of these units to BFN, it is reasonable to conclude that the response seen at these units would be comparable to that which would be seen at BFN.

The EPU test program has been evaluated and determined to adequately demonstrate that the SSCs will perform satisfactorily in service. As discussed earlier, no modifications are to be performed as part of EPU implementation that would cause BFN to behave significantly different from previous operating experience. Transient experience for other operating BWR plants for a wide range of power levels has shown a close correlation of the plant transient data to the predicted response and response was bounded by the plant safety analyses. It can be concluded that large transients, either planned or unplanned, have confirmed predicted plant response to that of transient modeling.

b. <u>Introduction of New Thermal-Hydraulic Phenomena or Identified System Interactions</u>

As discussed earlier, no modifications are to be performed as part of EPU implementation that would cause BFN to behave differently from previous operating experience for turbine stop valve trip/turbine control valve trip event. Related to the valve closure, EPU will have no impact on the components important to achieving the desired thermal performance. RPS logic is unaffected and as discussed in PUSAR Section 2.5.1, overall control rod insertion times will not be significantly affected. The event does not produce any unique operating situations that have not been demonstrated previously by operation of the MSRVs, turbine bypass valves, EHC, feedwater and feedwater control systems and there is no functional dependency between the closure of the turbine stop/control valves and the successful operation of these systems. As discussed in the PUSAR, Section 3.1 the MSRV setpoints are increased consistent with the operating dome pressure increase. The uprated nominal MSRV setpoints ensure that adequate differences (simmer margin) between operating pressure and MSRV setpoints are maintained. Therefore, there is no effect on valve functionality (opening/closing). Transient analyses have confirmed that the overall system responses and thermal-hydraulic phenomena are not affected by operation at EPU conditions. Therefore, there are no new thermal-hydraulic or identified system interactions.

c. <u>Facility Conformance to Limitations Associated With Analytical Analysis Methods</u>

The safety analyses performed for BFN used NRC-approved transient modeling codes. The NRC has accepted these for BWRs with a range of power levels and power densities that bound the requested power uprate for BFN. The codes have been benchmarked against BWR test data and have incorporated industry experience gained from previous transient modeling. Analyses use plant specific inputs and models all the essential physical phenomena for predicting integrated plant response to the analyzed transient. Thus, the codes will accurately and/or conservatively predict the integrated plant response to this transient at EPU power levels and no new information about transient modeling is expected to be gained from performing this large transient test.

d. <u>Plant Staff Familiarization With Facility Operation and Trial Use of Operating and Emergency Operating Procedures</u>

As discussed in Section 10.6 of the PUSAR, some additional training is required to enable plant operation at EPU conditions and power level. For EPU conditions, the scope of operator responses to transient, accident and special events are not affected.

e. <u>Margin Reduction in Safety Analysis Results for Anticipated Operational Occurrences</u>

The limiting pressurization transient events at EPU are the MSIV closure and turbine trip with turbine bypass failure. Both events are analyzed with failure of direct scram (i.e., scram based on neutron flux instead of valve position). The BFN turbine trip analyses assume that the events initiate at 102% of EPU reactor thermal power, the reactor dome pressure is 1055 psig, one MSRV (with the lowest setpoint) is out-of-service (OOS) and uses BFN turbine stop valve closure times. The results of the EPU overpressure protection analysis for the turbine trip event are given in Figure 3-2 of the PUSAR. The peak EPU pressure for this transient is 1298 psig versus a peak pressure of 1342 psig for MSIV closure. This analysis demonstrates that the MSIV closure transient described above remains the limiting event in terms of margin to reactor pressure vessel peak pressure.

No new information with regard to transient modeling or the analysis results are expected to be gained from performing this large transient test. Therefore TVA concludes that performance of a turbine stop/control valve closure transient test would not yield sufficient information to warrant conducting this test.

f. Guidance Contained in Vendor Topical Reports

The EPU license application was prepared following the guidelines contained in the NRC approved General Electric (GE) Company Licensing Topical Reports NEDC 32424P-A (ELTR1), February 1999, and NEDC 32523P-A (ELTR2), February 2000, and its Supplement 1, Volumes I and II. Appendix L, Section L.2.4 of ELTR1 currently discusses the need to perform large transient testing, specifically a Main Steam Isolation Valve (MSIV) closure test and a Generator Load Rejection test. Therefore, as indicated for each initial startup test in Table 1, a BFN plant-specific basis for either performing or taking exception to re-performing the original startup large transient tests is provided.

g. Risk Implications

While this application is not "risk-informed," TVA believes, on a qualitative basis, that the benefits from performing these tests are negligible, when assessed against the risks of subjecting the plant to an otherwise unnecessary challenge. A scram from high power level results in an unnecessary and undesirable transient cycle on the primary system. In addition, the risk posed by intentionally initiating a large transient event, although small, should not be incurred unnecessarily.

D. Evaluate the Adequacy of Proposed Transient Testing Plans

1. General Discussion

BFN will approach the power increase associated with EPU in a controlled manner using testing that will be used to demonstrate that the plant operates within design parameters.

2. Specific Acceptance Criteria

BFN does not propose to utilize any new types of testing to demonstrate performance at EPU conditions. BFN also does not intend to intentionally initiate a large transient test at high power such as a main steam isolation valve closure or turbine trip. The testing that will be performed will be based upon standard plant procedures such as post modification testing, Technical Specification surveillance testing and required startup tests. The approach to EPU conditions from current licensed power level will be conducted in a stepped fashion with appropriate holds as shown in Table 1 for evaluation of test data. The proposed testing will ensure the plant parameters respond as expected to various system perturbations.

In the event a plant system does not respond as expected, the test will be put on hold and the plant maintained in a safe condition until the issue is resolved. This is a standard practice and requires no special controls related to EPU.

By conducting the power ascension in a controlled, step fashion, the time a plant system is in an untested configuration above the current licensed power level is minimized. The EPU-related testing is expected to be conducted in an expeditious manner during restart from a refueling outage as permitted by grid and/or environmental conditions. Thus emphasis will be placed on the timely completion of testing activities. The EPU ascension will include the required Technical Specification testing and Quality Assurance requirements.

IV. CONCLUSION

TVA‘s evaluation has confirmed that the planned scope of EPU testing coupled with the BFN Unit 1 Restart Testing Program is adequate and additional large transient testing is neither required nor prudent. Steady state testing confirms the important nuclear characteristics required for transient analyses. Technical Specification required surveillance testing (e.g., component testing, trip logic system testing, simulated actuation testing) demonstrates that the systems, structures and components (SSCs) will perform their functions, including integrated performance for transient mitigation as assumed in the transient analyses. The characteristics and functions of SSCs do not need to be demonstrated further in a large transient test. In addition, the limiting transient analyses (i.e., those that affect core operating and safety limits) are re-performed each cycle and are included as part of the reload licensing analysis.

Large transient testing is normally performed on new plants because experience does not exist to confirm a plant's operation and its response to transients. However, these initial tests are not normally performed for plant modifications following initial startup because of the well-established quality assurance and maintenance programs including component and system level post modification testing. Large transient tests only challenge a limited set of systems and components. This situation results because the plant would respond to the transient using appropriate safety and non-safety systems and thus the plant response would be much milder than the limiting transients analyzed in the UFSAR. Also, it would not be expected that a limiting single failure would occur randomly in conjunction with the test. This situation of only challenging a limited number of systems also results because the transient is rapidly mitigated and the long term consequences are relatively benign and controlled by normal operator action. Actuation of a safety system in the long term phases of such an event typically does not occur until the reactor system conditions have returned to near normal conditions. In this near normal condition, the actuation closely resembles that which would occur during normal surveillance testing. No new significant information would be gained by imposing a large transient as opposed to surveillance testing and thus large transient tests impose an undue strain on the plant systems without sufficient return of information.

Based on the (a) similarity of the BFN design configuration and system functions at pre-EPU to post-EPU; (b) results of industry EPU experience and responses to unplanned transients; (c) the fact that past transient and safety analyses results correlate closely to results from actual transients; and, (d) the evaluation of unplanned transients for the pre-EPU BFN and other post-EPU plants that provide favorable comparison of plant responses, it is reasonable and justifiable that the effects at EPU conditions can be determined by existing requirements of the Technical Specifications and plant procedures. For large transients, the effect at EPU conditions can be analytically determined on a plant specific basis versus actual transient testing. The transient analyses performed for the BFN EPU demonstrate that all safety criteria are met and that EPU does not cause any previous non-limiting events to become limiting. No safety related systems will be significantly modified for the EPU. Some instrument setpoints were changed but the setpoint changes themselves do not measurably contribute to the response to large transient events. The associated post modification testing will confirm proper response of the equipment.

As has been shown, analyses for EPU provide the necessary assurance that sufficient margins to safety limits are maintained. Should any future large transients occur, BFN procedures require verification that the actual plant response is in accordance with the predicted response. Plant event data recorders are capable of acquiring the necessary data to confirm the actual versus expected response. The important nuclear characteristics required for transient analysis are confirmed by the steady state physics testing. Transient mitigation capability is demonstrated by equipment surveillance tests required by the Technical Specifications. In addition, the limiting transient analyses are included as part of the reload licensing analysis.

A scram and isolation from high power is an undesirable transient cycle on the primary system. Because past testing at BFN and evaluation of pre-EPU operational experience at BFN and post-EPU experience at other plants has shown that large transient plant responses are within the bounds of plant transient analyses, additional large transient testing involving scram from high power is not justified or necessary.

V. REFERENCES

1. TVA letter, T. E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) – Unit 1 - Proposed Technical Specifications (TS) Change TS - 431- Request for License Amendment - Extended Power Uprate (EPU) Operation," dated June 28, 2004.

2. TVA letter, TE Abney to NRC, "Browns Ferry Nuclear Plant (BFN) – Unit 1 Response to NRC's Acceptance Review Letter and Request for Additional Information Related to Technical Specifications (TS) Change No. TS-418, Request for Extended Power Uprate Operation," dated February 23, 2005.

3. NRC Letter to TVA, "Browns Ferry Nuclear Plant Units 1 and 3 – Restart Test Program," dated August 30, 1994.

4. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Unit 1 Regulatory Framework for the Restart of Unit 1," dated December 13, 2002.

5. NRC Letter to TVA, "Regulatory Framework for the Restart of Browns Ferry Nuclear Plant Unit 1," dated August 14, 2003.

6. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Restart Test Program (RTP) for Units 1 and 3," dated September 27, 1991.

7. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Request for Additional Information Regarding the Restart Test Program for Units 1 and 3," dated February 18, 1992.

8. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Update of Restart Test Program (RTP) for Units 1 and 3," dated December 28, 1992.

9. TVA letter to NRC dated July 19, 1993, "Browns Ferry Nuclear Plant (BFN) – Restart Test Program (RTP) Update for Units 1 and 3."

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
STP 1	Chemical and Radiochemical: Before fuel loading, a complete set of chemical and radiochemical samples were taken to ensure that all sample stations were functioning properly and to determine initial water quality. Subsequent to fuel loading during reactor heatup and at major power level changes, samples were taken and measurements made to determine the chemical and radiochemical quality of reactor water and reactor feedwater, amount of radiolytic gas in the steam, gaseous activities leaving the air ejectors, decay times in the offgas lines, and performance of filters and demineralizers. Calibrations were made of monitors in the stack, liquid waste system, and liquid process lines.	Y	Y	Y	Open Vessel Testing, Initial Heatup, Power Testing	Yes (Standard plant procedure)	The purpose of the original test included (a) securing information and knowledge about the quality of the reactor coolant chemistry, (b) determination that the sampling equipment, procedures, and analytic techniques are adequate to supply the data required to demonstrate that the coolant chemistry met water quality specification and process requirements, and (c) evaluation of fuel performance, operation of demineralizers and filters, operation of the offgas system and calibration of certain process instruments. For test purpose (a), EPU testing will include sampling and measurements at selected power levels to determine 1) the chemical and radiochemical quality of reactor water and feedwater and 2) gaseous release. For test purposes (b) and (c), the current Browns Ferry chemistry and plant performance monitoring programs gather information on plant equipment and system performance. This information is evaluated in order to maintain equipment, system and plant performance within process requirements, chemistry/radiochemistry specifications and guidelines and fuel warranty. The demonstration of the sampling equipment, procedures, analytic techniques, and operation validation		X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							need not be repeated. Therefore, this purpose of the original testing is not applicable to EPU and is not required.							
STP 2	Radiation Measurements: A survey of natural background radiation throughout the plant site was made before fuel loading. Subsequent to fuel loading, during reactor heatup and at power levels of 25, 50, and 100 percent of rated power, gamma radiation level measurements and, where appropriate, thermal and fast neutron dose rate measurements were made at significant locations throughout the plant. All potentially high radiation areas were surveyed.	Y	Y	Y	Open Vessel Testing, Initial Heatup, Power Testing	Yes (EPU startup test)	The purpose of the original test was (a) to determine the background radiation levels in the plant environs prior to operation for base data on activity buildup and (b) to monitor radiation at selected power levels to assure the protection of personnel during plant operation. For test purpose (a), the demonstration of background radiation levels need not be repeated. Therefore, this purpose of the original testing is not applicable to EPU and is not required. For test purpose (b), EPU testing at selected EPU power levels will take gamma dose measurements and where appropriate, neutron dose measurements at specific limiting locations throughout the plant to assess the impact of the uprate on actual plant area dose rates.		X	X	X	X	X	X
STP 3	Fuel Loading: Before fuel loading, control rods were installed and tested. A neutron source of approximately 10 neutrons per sec was installed near the center of the core. At least	Y	Y	Y	Open Vessel Testing	Yes (Standard plant procedure)	The purpose of the original test was to load fuel safely and efficiently to the full core size. Current Technical Specifications and approved plant procedures will effectively govern the safe and efficient loading of fuel for EPU implementation. The normal	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	three neutron detectors calibrated and connected to high flux scram trips were located to produce acceptable signals during loading. Fuel Loading was begun at the center of the core and proceeded radially to the fully configuration. As each cell was loaded, the following checks were performed: 1) Subcriticality check, 2) Control Rod Function test, 3) Fuel Loading check, 4) Repeat of Subcriticality check, and 5 Repeat of the Control Rod Function test. Shutdown margin demonstrations were performed periodically during fuel loading.						refueling test program for open vessel testing will accomplish required testing and verification. During refueling, each fuel movement is verified to a pre-determined fuel assembly transfer form and assembly movement and subcritical multiplication is monitored. Upon completion of core loading, a final verification of fuel assembly loading and orientation is completed. During refueling, subcriticality checks are performed. Control rod function (withdrawal and insertion) is completed with the core fully loaded. These procedures and actions meet the intent of the original testing.							
STP 4	Full Core Shutdown Margin: This test was performed in the fully loaded core at ambient temperature in the xenon-free condition. Subcriticality was demonstrated with the strongest rod fully withdrawn and a series of calibrated rods pulled to a position calculated to be equal to a shutdown margin specified to account for expected reactivity changes	Y	Y	Y	Open Vessel	Yes (Standard plant procedure)	The purpose of the original testing was to demonstrate that the reactor would be subcritical throughout the first fuel cycle with any single control rod withdrawn. The core shutdown margin requirement is not changed by EPU. Shutdown margin testing is performed during each refueling before any fuel handling over an open reactor vessel and the normal refueling test program for heatup will accomplish required shutdown margin testing for EPU. These procedures and actions meet the	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	during core lifetime.						purpose of the original testing.							
STP 5	Control Rod Drive System: The CRD tests performed verified that all control rod drives operated properly when installed and also periodically during heatup to assure that there is no significant binding caused by thermal expansion of the core components.	Y	Y	Y	Open Vessel Testing, Initial Heatup, Power Testing	Yes (Standard plant procedure)	The purpose of the original test was to (a) demonstrate that the CRD system operated properly over the full range of primary coolant temperature and pressures from ambient to operating and particularly that thermal expansion of core components does not bind or significantly slow control rod movements, and (b) to determine the initial operating characteristics of the entire CRD system. For test purpose (a), as stated in Section 2.5.1 of the EPU safety analysis report, no change is made to the control rods due to the EPU and the scram times are decreased by the transient pressure response. The normal refueling test program will accomplish control rod drive system testing and troubleshooting and CRDM scram insertion timing for EPU implementation. For test purpose (b), as stated in Section 2.5.2 of the EPU safety analysis report, the CRD positioning and cooling functions are not affected by EPU. Confirmation that the system meets TS requirements for operability for startup and power ascension is required by the refueling test program. As the EPU does not have an effect on the CRD System, the current verification per the refueling test program is valid for EPU operations.	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
STP 6	SRM Performance and Control Rod Sequence: The operational neutron sources were installed and source range monitor count rate data taken during rod withdrawals to critical and compared with stated criteria on signal and signal count-to-noise count ratio. Control Rod patterns were recorded periodically as the reactor was heated to rated temperature. As each rod group was completed during the power ascension, the electrical power, steam flow, and APRM response were recorded. Movement of rods in a prescribed sequence was monitored by the Rod Worth Minimizer and Rod Sequence Control System to prevent unacceptable our-of-sequence control rod movements during startup or shutdown.	Y	Y	Y	Open Vessel Testing, Initial Heatup, Power Testing	Yes (Standard plant procedure)	The purpose of the original testing was to demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner. It also determined the effect of typical rod movements on reactor power. This was an initial startup test requirement with regard to startup neutron sources to achieve initial criticality in a safe and efficient manner for the control rod withdrawal sequences. Operation at EPU increases the upper end of the power operating domain. These changes in the higher end do not significantly or directly affect the manner of operating or response of the reactor in the startup/low power range. The neutron monitoring is confirmed to be operating properly at low power via standard plant procedures/processes The rod worth minimizer controls rod patterns to ensure compliance with the prescribed rod patterns. Therefore, additional SRM performance testing is not required for EPU	NA						
STP 9 (Units 2/3 only).	Water Level Measurement: The first part of testing measured the YARWAY	N	Y	Y	Initial Heatup, Power	Yes (Standard plant	The purpose of the original testing was to a) verify the calibration and b) verify the agreement of the various narrow and wide range level indicators under various		X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
See STP 39 for Unit 1)	reference to verify agreement with the temperature correction factor used in calibration. The second part of the test consisted of determining the agreement of the water level instrumentation at two core flow rates and various heights.				Testing	procedure)	conditions. As part of the restart efforts, the Yarway temperature equalizing columns are removed and the instruments replaced with analog trip system devices. The hardware associated with this new reactor water level instrumentation is not modified and normal operational water level and level setpoints (alarms/trips/ actuations) are not changed by EPU. The demonstration of procedures and operational validation for EPU therefore need not be repeated.							
STP 10	IRM Calibration/Performance: Initially, the IRM system was set to maximum gain. After the APRM heatup calibration and after the first heat balance calibration of the APRM's, the IRM-APRM, overlap was checked and the IRM gains adjusted if necessary to improve the IRM system overlap between the SRMs and IRMs.	Y	Y	Y	Open Vessel Testing, Initial Heatup, Power Testing	Yes (EPU startup test)	The purpose of the original testing was to adjust the IRM system to obtain an optimum overlap with the SRM and APRM systems. The IRM overlap with the SRMs is not affected by EPU. The APRMs will be re-referenced to read 100% at EPU conditions. After the APRM calibration for EPU, the IRM gains will be adjusted as necessary to assure the IRM overlap with the APRMs. This will be performed during the first controlled shutdown following APRM calibration for EPU.	(Not a startup test, but will be done during first controlled shutdown following APRM calibration for EPU)						
STP 11	LPRM Calibration: The LPRM channels were calibrated to make the LPRM readings proportional to the	Y	Y	Y	Power Testing	Yes (Standard plant procedure)	The purpose of the original testing was to calibrate the LPRM system. LPRM calibration is performed periodically as specified in the Technical	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	average heat flux in the four corner fuel rods surrounding each chamber at the chamber elevation. The initial calibration factors were obtained from measurements of axial power distribution, precalculated local power distributions, and precalculated radial power distributions.						Specifications using approved plant procedures. The normal refueling test program will accomplish APRM calibration during startup to EPU in accordance with the refueling test procedure. The method and approach used to perform LPRM calibration is not affected by EPU. These procedures and actions will meet the intent of the original testing.							
STP 12	APRM Calibration: The APRM's were initially adjusted to maximum amplifier gain. A low power calibration was performed based on heat balance calculations during reactor heatup. When reactor power was high enough to obtain more accurate steady state heat balances to determine actual core thermal power, the APRM channels were calibrated to read percent of core thermal power.	Y	Y	Y	Power Testing	Yes (Standard plant procedure)	The purpose of the original testing was to calibrate the APRMs. APRMs will be calibrated in accordance with Technical Specifications during startup from each refueling outage by the normal refueling test program. For EPU, each APRM channel will be adjusted to be consistent with the core thermal power, referenced to the EPU level as determined from the heat balance. These procedures and actions will meet the intent of the original testing.	X						
STP 13	Process Computer: Following fuel loading, during plant heatup, and the ascension to rated power, the nuclear steam supply system	Y	Y	Y	Prior to Startup, Open Vessel Testing, Initial Heatup,	Yes (Standard plant procedure)	The purpose of the original testing was to verify the performance of the process computer under plant operating conditions. Plant process computer data installation		X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	and the balance of plant system process variables sensed by the computer became available. As station process variable signals became available to the computer, verification was made that the computer was receiving correct values of sensed process variables and that the results of performance calculations of the nuclear steam supply system and the balance-of-plant were correct.				Power Testing		and verification is confirmed during startup from each refueling outage. The normal refueling test program will accomplish this testing during startup and power ascension at EPU. This testing will meet the intent of the original testing.							
STP 14	RCIC System: Flow tests of the RCIC System were performed at reactor pressures between 150 and 1,020 psig. These tests were designed to verify proper operation of the RCIC System, determine time to reach rated flow and adjust flow controller in RCIC System for proper flow rate. These tests were first performed with the system in the test mode so that discharge was not routed to the reactor vessel and the final demonstration routed discharge flow to the reactor vessel while the reactor was at	Y	Y	Y	Initial Heatup, Power Testing	Yes (Standard plant procedure)	The purpose of the original testing was to verify the proper operation of the RCIC system over its required operating pressure range. Unit 2/3 - As discussed in EPU safety analysis report Sections 3.8 and 9.1.3 there is no change to the normal reactor operating pressure and the MSRV setpoints remain the same compared to those associate with the 105% power uprate and the associated 30 psi pressure increase. For operation at EPU, there is no change to the maximum specified reactor pressure for RCIC System operation, no changes to the RCIC System performance parameters, and no effect on the maximum reactor pressure postulated to be present during system startup. Current testing is	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	partial power.						performed by testing with suction from the condensate storage tank with discharge back to the condensate storage tank. Therefore, no changes are required to meet the performance requirements for the RCIC System or to limit the maximum startup transient speed peak. RCIC System testing, including automatic starts from cold conditions is governed by Technical Specifications and approved plant procedures. During unit startup, the 150 psig test must be completed and RCIC declared operable. As EPU does not have an effect on the RCIC System, the current testing per the unit startup procedures is valid for EPU operations. Therefore, additional specific system testing at EPU conditions is not required. Unit 1 - During unit startup, RCIC reliability will be demonstrated by cold quick starts with the pump aligned in the normal EPU operating reactor pressure range. After the auto start portion of the test while the system is in operation, small step disturbances in flow command will be input to demonstrate satisfactory turbine control stability. Testing will be performed by testing with suction from the condensate storage tank with discharge to the reactor vessel. Additional specific system testing at EPU conditions is not required.							

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
STP 15	HPCI System: Flow tests of the HPCI System were performed at reactor pressures between 150 and 1,020 psig. These tests were designed to verify proper operation of the HPCI system, determine time to reach rated flow, and adjust the flow controller in HPCI system for proper flow rate. These tests were first performed with the system in the test mode so that discharge was not routed to the reactor vessel and the final demonstration routed discharge flow to the reactor vessel while the reactor was at partial power.	Y	Y	Y	Initial Heatup, Power Testing	Yes (Standard plant procedure)	The purpose of the original testing was to verify the proper operation of the HPCI system over its expected operating pressure range. Unit 2/3 - As discussed in the EPU safety analysis report Sections 4.2.1 and 4.3, there is no change to the maximum specified reactor pressure for HPCI system operation, no changes to the HPCI system performance parameters, and no effect on the maximum reactor pressure postulated to be present during system startup compared to those associate with the 105% power uprate and the associated 30 psi pressure increase. For operation at EPU, no changes are required to meet the performance requirements for the HPCI system or to limit the maximum startup transient speed peak. HPCI System testing, including automatic starts from cold conditions is governed by Technical Specifications and approved plant surveillance procedures. During unit startup, the 150 psig test must be completed and HPCI declared operable. As EPU does not have an effect on the HPCI System, the current testing per the unit startup procedures is valid for EPU operations. Current testing is performed by testing with suction from the condensate storage tank with discharge back to the condensate storage tank.. Therefore, additional specific system	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							testing at EPU conditions is not required. Unit 1 - During unit startup, HPCI reliability will be demonstrated by cold quick starts with the pump aligned in the normal EPU operating reactor pressure range. After the auto start portion of the test while the system is in operation, small step disturbances in flow command will be input to demonstrate satisfactory turbine control stability. Testing will be performed by testing with suction from the condensate storage tank with discharge to the reactor vessel. Additional specific system testing at EPU conditions is not required.							
STP 16	Selected Process Temperatures: Applicable reactor parameters were monitored during the initial heatup, the initial cooldown, and after recirculation pump trips in order to determine that adequate mixing of the reactor water was occurring in the lower plenum of the pressure vessel. The adequacy of the bottom-drain-line thermocouple as a measure of bottom reactor vessel temperature was also determined.	Y	Y	Y	Power Testing	No (Low power condition not affected by EPU)	The purpose of the original testing was (a) to establish the proper setting for the low speed limiter for the recirculation pumps and (b) to provide assurance that the measured bottom head drain temperature corresponds to the bottom head coolant temperature during normal operations. The original test obtained RPV temperatures during rapid heatup and cooldown to confirm thermal analysis models. EPU does not significantly affect RPV temperatures during rapid heatup or cooldown. Since thermal analysis models were confirmed during the initial startup testing and RPV temperatures are not significantly affected during EPU, this	NA						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							testing is not required for EPU conditions.							
STP 17	System Expansion: System expansion checks were made of major equipment and piping in the nuclear steam supply system during heatup to assure components are free to move as designed and adjustments were made as necessary for freedom of movement.	Y	Y	Y	Initial Heatup, Power Testing	No (Unit 2/3) (Low power condition not affected by EPU) Yes (Unit 1) (EPU startup test)	The purpose of the original testing was to (a) verify the reactor drywell piping was free and unrestrained in regard to thermal expansion, (b) verify that suspension components were functioning as required, and (c) provide data for calculation of stress levels in nozzles and weldments. Unit 2/3 - Full system expansion due to thermal effects is experienced at low power conditions and does not increase in proportion to power level. Since EPU does not include a reactor vessel pressure increase nor a corresponding primary coolant temperature increase, the thermal expansion of drywell piping is not affected by EPU conditions. Therefore, this test is not required for EPU. Unit 1 – Due to the 30 psi reactor pressure increase (and associated temperature increase), system expansion checks will be made for major equipment and piping in the nuclear steam supply system during heatup to assure components are free to move as designed and adjustments will be made as necessary for freedom of movement.	X						
STP 18	Core Power Distribution:	Y	Y	Y	Power	Yes (Standard	The purpose of the original testing was to (a) confirm the reproducibility of the TIP		X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	Core Power distribution, including power symmetry, was obtained during the power ascension program. Axial power traces were obtained at each of the TIP locations. The results of the complete set of TIP traces were evaluated to determine core power symmetry.				Testing	plant procedure)	system readings, (b) determine the core power distribution in three dimensions, and (c) determine core power symmetry. There are no changes to the TIP system as a result of the EPU. TIP data and Core Monitoring System data are taken and analyzed to determine TIP asymmetry and core power symmetry during each startup refueling test program in accordance with Technical Specifications and approved plant procedures. This testing meets the intent of the original testing.							
STP 19	Core Performance: Unit 1 & 2 - Core power level, maximum heat flux, recirculation flow rate, hot channel coolant flow, MCHFR, fuel assembly power, and steam qualities were determined at existing power levels. Plant and incore instrumentation, conventional heat balance techniques and core performance worksheets and nomograms were used. This was performed above 10 percent power and at various pumping conditions and independent of the process computer functions. Unit 3 – Core parameters were	Y	Y	Y	Initial Heatup, Power Testing	Yes (EPU startup test)	The purpose of the original testing was to (a) evaluate the core thermal power and (b) evaluate core performance parameters of Maximum Linear Heat Generation Rate (MLHGR), Minimum Critical Power Ratio (MCPR), and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR). Core performance parameters will be calculated during EPU to verify they remain within limits as part of a careful, monitored approach to the maximum EPU power level. This monitored approach to EPU power levels meets the intent of the original testing.		X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	evaluated by manual calculations, the process computer, or the off-line computer program BUCLE.													
STP 20	Electrical Output and Heat Rate (U1/U2): Steam Production (U3): The plant gross electrical output was measured during sustained operation (maintained for 300 hours) at rated conditions to demonstrate that the guaranteed gross electrical output requirements were satisfied without exceeding the reactor power level warranty and to determine a preliminary net plant heat rate value. For Unit 3 only, the steam production rate was measured during two 2-hour periods at conditions prescribed in the Nuclear Steam Generating System warranty.	Y	Y	Y	Power Testing	Yes (EPU startup test)	Units 1, 2, 3 - The purpose of the original testing was to demonstrate that the plant net electrical output and net heat rate requirements are satisfied. This test was used to demonstrate reactor vendor warranty requirements and does not pertain to the safe operation of the plant. Plant electrical output versus reactor thermal power will be closely observed during power ascension to determine plant heat rate.							X
STP 21	Flux Response To Rods: Rod movement tests were made at chosen power levels to demonstrate that the transient response of the	Y	Y	Y	Power Testing	Yes (Standard plant procedure)	The purpose of the original testing was to demonstrate the stability of the core local power-reactivity feedback mechanism with regard to small perturbations in reactivity caused by rod movement.		X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	reactor to a reactivity perturbation was stable for the full range of reactor power. A centrally located rod was moved and the neutron flux signal from a nearby LPRM chamber measured and evaluated to determine the dynamic effects of rod movement.						Reactor core stability monitoring is performed continuously by a dedicated system and by Operations department oversight. Core stability monitoring is performed without moving control rods. Analytical stability evaluations are core reload dependent and are performed for each reload fuel cycle. The combination of monitoring and evaluations fulfills the intent of the original startup test.							
STP 22	Pressure Regulator: The pressure setpoint was decreased rapidly and then increased rapidly in steps and the response of the system was measured. The backup regulator was tested by increasing the operating pressure regulator setpoint rapidly until the backup regulator took control. The load reference setpoint was reduced, and the test repeated with the bypass valves in control. The response of the system was measured and evaluated and regulator settings optimized.	Y	Y	Y	Initial Heatup, Power Testing	Yes (EPU startup test)	The purpose of the original testing was to (a) determine the optimum settings for the pressure control loop by analysis of the transients induced in the system by means of the pressure regulators, (b) demonstrate the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator and to set spacing between the setpoints at an appropriate value, and (c) demonstrate smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam used by the turbine. The original EHC system has been modified to incorporate a digital control system with redundant channels which eliminates the need for a backup regulator. As discussed in EPU safety analysis report Section 5.3.13, pressure control operational testing will be performed during EPU power ascension.	X	X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							For each defined test condition, the pressure control system response to pressure setpoint testing will be evaluated. EPU safety analysis report Section 5.3.11 also discusses that EPU startup ascension test or normal plant surveillance will be used to validate the TSV/TCV scram bypass interlock							
STP 23	Feedwater System: U1/U2/U3 - Reactor water level setpoint changes were used to evaluate and adjust the feedwater control system settings for all power and feedwater pump modes. U1 - Each feedwater pump was operated through its flow range to verify acceptable pump linearity. Response time on each feedwater pump was verified by changing the flow by 10 percent and measuring the turbine speed and flow response times. U2/3 – One of the three operating feedwater pumps was tripped and the automatic flow runback circuit acted to drop power to within the capacity of the remaining pumps.	Y	Y	Y	Power Testing	Yes (EPU startup test)	The purpose of the original testing was to (a) adjust the feedwater control system for acceptable reactor water level control, (b) demonstrate stable reactor response to subcooling changes, and (c) demonstrate the capability of the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump. For purposes (a) and (b), as discussed in EPU safety analysis report Section 5.2.2, control system tests will be performed at the appropriate plant conditions for that test at each of the power increments, to show acceptable adjustments and operational capability. The feedwater level control system will be tuned via post modification testing to account for the increased EPU pumping capacity of the feedwater system. The feedwater level control system is capable of controlling the reactor vessel level to prevent both level increases or decreases such that no new ECCS trips or actuations will occur based on vessel level. Therefore, this		X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							post modification testing will ensure proper operation of the system. For purpose (c), the original Feedwater System startup testing included a feedwater pump trip test. EPU safety analysis report Section 9.1.3 states that the loss of one feedwater pump event only addresses operational considerations to avoid reactor scram on low reactor water level (Level 3). This requirement is intended to avoid unnecessary reactor shutdowns. This capability will be verified at a high reactor power condition. Based upon the upgrades to the condensate, condensate booster, and feedwater pumps, it is not expected that a recirculation pump runback will occur during this test.							
STP 24	Bypass Valves: One of the turbine bypass valves was tripped open and closed. The pressure transient was measured and evaluated to aid in making final adjustments to the pressure regulator.	Y	Y	Y	Power Testing	Yes (Standard plant procedure)	The purpose of the original testing was to (a) demonstrate the ability of the pressure regulator to minimize the reactor pressure disturbance during an abrupt change in reactor steam flow and (b) demonstrate that a bypass valve can be tested for proper functioning at rated power without causing a high flux scram. As stated in EPU safety analysis report Section 5.2.1.1, no modifications to the turbine control valves or the turbine bypass valves are required for operation at the EPU conditions. Confirmation testing will be performed during power	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							operation.							
STP 25	Main Steam Isolation Valves: Fast full closure of each MSIV was performed at hot standby and selected power levels to determine a) the maximum power conditions at which individual valve full closure tests could be performed without a reactor scram, b) functional checks (10 percent closure) of each isolation valve were performed at selected power levels above the maximum power condition for individual MSIV full closure, c) a test of simultaneous full closure of all MSIV's was performed at about 100 percent of rated thermal power and proper operation of the relief valves and the RCIC were shown, reactor process variable were monitored to determine the transient behavior of the system during and following each isolation test d) MSIV delay and movement times were determined and proper seating of the MSIVs demonstrated.	Y	Y	Y	Initial Heatup, Power Testing	Not applicable (Part a & b) No (Part c) Yes (Part d) (Standard plant procedure)	For purposes (a) and (b), the percent power level at which a single MSIV can be closed without a scram may change due to EPU instrument-related changes. However, determination of this power level is a plant capacity consideration rather than a demonstration of proper operation of a system(s) and this testing is not a requirement to safely implement EPU. For purpose (c), a simultaneous full closure of all MSIVs is a large transient test. See Section III.C for a discussion of test performance. For purpose (d), valve movement (i.e. stroke) will be verified via a standard plant procedure. Proper seating of the valves will be conducted via performance of the Appendix J local leak rate testing.	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
STP 26	Relief Valves: The main steam relief valves were each opened manually. Capacity of each relief valve was determined by the amount the bypass or control valves closed to maintain reactor pressure. Proper reseating of each relief valve was verified by observation of temperatures in the relief valve discharge piping. U2/3 addition – At selected test conditions, each valve was manually actuated and at least one valve was timed. U2 addition - Additional timing data was obtained in conjunction with those transient tests which result in automatic relief valve opening.	Y	Y	Y	Power Testing	Yes (Standard plant procedure)	The purpose of the original testing was to (a) verify the proper operation of the primary system relief valves, (b) determine the capacity and response characteristics of the relief valves, and (c) verify proper seating of the relief valves following operation. Unit 2/3 - For purposes (a) and (c), the Main Steam Relief Valve Manual Cycle Test is performed once per operating cycle in accordance with Technical Specifications and approved plant procedures. During unit startup from refueling, each MSRV is verified to open and close when manually actuated at rated reactor pressure. As described in EPU safety analysis Section 3.1, no MSRV setpoint increase is needed because there is no change in the dome pressure or simmer margin. Therefore, there is no effect on valve functionality (opening/closing). Therefore, Technical Specification testing of MSRV operation is considered appropriate for EPU. For purpose (b), the data that was collected regarding relief valve capacity is still valid for the installed valves and is not impacted by operation at EPU conditions. Therefore it is not necessary to repeat this portion of the original test. Unit 1 – As part of EPU implementation	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							and the associated 30 psi pressure increase, the main steam relief valves will be reset to the higher set pressures. For purposes (a) and (c), the Main Steam Relief Valve Manual Cycle Test is performed once per operating cycle in accordance with Technical Specifications and approved plant procedures. . During unit startup from refueling, each MSRV is verified to open and close when manually actuated at rated reactor pressure. For purpose (b), the data that was collected regarding relief valve capacity is still valid for the installed valves and is not impacted by operation at EPU conditions. Therefore it is not necessary to repeat this portion of the original test.							
STP 27	Turbine Stop and Control Valve Trips (U1): Turbine Trip and Generator Load Rejection (U2/3): Unit 1-The stop or control valves were tripped closed at selected reactor power levels (50% and 100%). Neutron flux, feedwater flow and temperature, vessel water level and pressure were monitored. Responses of selected control valves, stop valves, relief valves, and bypass valves were recorded. The ability to	Y	Y	Y	Power Testing	No	A turbine stop/control valve closure or generator load rejection at high power is a large transient test. See Section III.C for a discussion of test performance. The ability to ride through a load rejection within bypass capacity without a scram is performed at a low power level and is thus not affected by operation at EPU and therefore this test is not required to be re-performed.	NA						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	ride through a load rejection within bypass capacity without a scram was demonstrated at low power (25%). Unit 2/3 –The turbine stop valves were tripped at selected reactor power levels (50% and 100%) and the main generator breaker tripped in such a way that a load imbalance occurred. Several reactor and turbine operating parameters were monitored to evaluate the response of the bypass valves, relief valves, and reactor protection system. The peak values and change rates of reactor steam pressure and heat flux were determined. The ability to ride through a load rejection within bypass capacity without a scram was demonstrated at low power (25%).													
STP 29	Flow Control: Various process variables were recorded while step changes were introduced into the recirculation flow control system (increased and decreased) at chosen points on the 50, 75, and 100 percent	Y	Y	N	Power Testing	Yes (Standard plant procedure)	The original testing purpose was to (a) determine the plant response to changes in the recirculation flow, (b) optimize the setpoints of the flow controller, and (c) demonstrate the plant load following capability in the different flow control modes. Increased voids in the core during normal	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	load lines. Up to 30 percent/minute change in recirculation flow was made from all flow conditions down to the lower limit of the Master Flow Controller and return. Load following capability was demonstrated in all flow control modes. U2 – Ramp changes were made at rates within the range of 10 percent to 30 percent per minute. Load following capability was demonstrated in the automatic and master manual flow control modes.						EPU power operations requires a slight increase in recirculation drive flow to achieve the same core flow. Because adequate core flow can be maintained without requiring any changes to the recirculation system and only a small increase in pump speed for the same core flow, the response to flow changes will be similar to that displayed during the original startup testing. System modifications since the original plant configuration include a new flow control system and the installation of Variable Frequency Drives (VFDs) for the Reactor Recirculation System pump motors. Reactor Recirculation System testing and tuning of the current flow control system is performed during each refueling startup during vessel hydro conditions and also at power conditions to analyze system response to speed demand of small and large changes. The testing performed during the normal refueling test program will meet the intent of the original test objectives during startup and power ascension.							
STP 30	Recirculation System: U1/3-Single and both recirculation pumps were tripped at various power levels. Two pump trips were initiated by tripping the MG set drive	Y	Y	Y	Power Testing	Yes (Standard plant procedure)	The original testing purpose was to (a) evaluate the recirculation flow and power level transients following trips of one or both of the recirculation pumps, (b) to obtain recirculation system performance data, and (c) to verify that no recirculation system cavitation will occur on the	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	motors. One single pump trip at 50 percent power was initiated by opening the generator field breaker and the remaining single pump trips were initiated by tripping the MG set drive motor. Reactor operating parameters were recorded during the transient and at steady-state conditions. U1-The jet pump instrumentation was calibrated to read total flow. U1/2-MCHFR evaluations were made for conditions encountered during the transient. U2-With the recirculation pumps operating at the speed corresponding to rated flow at rated power, power was reduced by inserting rods to 23 percent power where the recirculation pumps would automatically run back to 20 percent speed and a check was made to determine if recirculation or jet pump cavitation occurred.						operable region of the power-flow map. Increased voids in the core during normal EPU power operations requires a slight increase in recirculation drive flow to achieve the same core flow. Because adequate core flow can be maintained without requiring any changes to the recirculation system and only a small increase in pump speed for the same core flow, the response to flow changes resulting from either a single or two pump trip will be similar to that of original startup testing. No additional testing is required for EPU. Verification of jet pump calibration is accomplished by standard plant procedures. See STP 35 for additional information. The verification of the non-occurrence of cavitation is not impacted by EPU since this condition only occurs at a low power/low flow condition. Therefore, it is not necessary to repeat this portion of the original testing.							
STP 31	Loss of Turbine-Generator and Offsite Power:	Y	Y	Y	Power Testing	Yes (Standard plant	The ability to mitigate a loss of offsite power is performed at a low power level and is thus not affected by operation at	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	The loss of auxiliary power test was performed at 25 percent (U2/3-20 to 30 percent) of rated power. The proper response of reactor plant equipment, automatic switching equipment, and the proper sequencing of the diesel generator loads were checked. Appropriate reactor parameters were recorded during the resultant transient.					procedure)	EPU and therefore this test is not required to be re-performed. The ability of the electrical equipment to respond to a loss of offsite power is demonstrated by testing required by the Technical Specifications for the onsite and offsite power distribution systems.							
STP 32	Recirculation M-G Set Speed Control (U1/2): Recirculation Speed Control and Load Following (U3): Several small step changes in speed demand were input at various pump speeds and appropriate recirculation loop transient signals recorded to demonstrate response performance over the full speed range with small speed demand step tests. (Note: The Recirculation M-G Sets replaced with Variable Frequency Drives.)	Y	Y	Y	Power Testing	Yes (Standard plant procedure)	The original testing purpose was to (a) determine the speed control characteristics of the MG sets in the recirculation control system, (b) obtain acceptable speed control system performance, and (c) determine the maximum allowable pump speed. This test determined the original as built characteristics of the Recirculation Control System. Increased voids in the core during normal EPU power operations requires a slight increase in recirculation drive flow to achieve the same core flow. Because adequate core flow can be maintained without requiring any changes to the recirculation system and only a small increase in pump speed for the same core flow, the response to flow changes will be similar to that of original startup testing.	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							System modifications since the original plant configuration include a new flow control system and the installation of Variable Frequency Drives (VFDs) for the Reactor Recirculation System Pumps. Reactor Recirculation System testing and tuning of the current flow control system is performed during each refueling startup during vessel hydro conditions and also at power conditions to analyze system response to speed demand of small and large changes. The testing performed during the normal refueling test program will meet the intent of the original test objectives during startup and power ascension at EPU.							
STP 33	Main Turbine Stop Valve Surveillance Test: At several test points, individual main turbine stop valves were closed. The response of the reactor was recorded and the maximum possible power level for performance of this test along with the 100 percent power flow control line established. Each stop valve closure was manually initiated and reset. Rate of valve stroking and timing of the close-open sequence was chosen to minimize the disturbance	N	Y	Y	Power testing	Yes (Standard plant procedure)	The original testing purpose was to demonstrate acceptable procedures for daily turbine stop valve surveillance tests at a power level as high as possible without producing reactor scram. Individual main turbine stop valves must be closed periodically during plant operation as required for plant surveillance testing. As described in EPU safety analysis report Section 3.5.2, the TSV bounding closing time was utilized in EPU analysis.	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	introduced.													
STP 34 (Unit 2/3 only) STP 90 (Unit 1 only)-	Vibration: Vibratory responses were recorded at various recirculation flow rates at temperatures below 150°F using strain gages on in-core guide tubes, control rod stub tubes, shroud support legs, and jet pump riser braces; accelerometers on the recirculation loops and displacement gages on the shroud, steam separator and jet pumps. Portable vibration sensor surveys were made on the recirculation loops and differential pressure measurements made across the core plates, shroud head and shroud wall. At hot, two-phase flow conditions, similar measurements were made on the in-core guide tubes, shroud, jet pump riser and shroud head. The results of vibration measurements made at other BWR installations will be considered in the final selection of components to be tested.	Y	Y	Y	Power testing	Yes (Standard plant procedure)	The original testing purpose was to obtain vibration measurements on various reactor components to demonstrate the mechanical integrity of the system to flow induced vibration and to check the validity and accuracy of the analytical vibration model. Previous startup tests obtained vibration measurements on various reactor pressure vessel internals to demonstrate the mechanical integrity of the system under conditions of flow induced vibration, and to check the validity of the analytical vibration model. With the exception of the steam dryer, the system flows associated with the reactor vessel internals are unchanged by operation at EPU conditions and thus additional testing is not required. The small increase in normal flow through the recirculation loops are within the values previously tested. Changes in the vibrations in the recirculation loops will be detected by the permanently installed recirculation pump vibration monitoring. Analysis of the reactor vessel internals at EPU power level was performed to ensure that the design continues to comply with the existing structural requirements. Results of this analysis are provided in EPU safety analysis	X	X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							Section 3.3 and are discussed in EPU submittal Enclosures 10 as follows: Flow induced vibration effects of the following components were evaluated per the requirements of NRC Regulatory Guide 1.20: Shroud, Jet Pumps, Jet Pump sensing lines, Steam Dryer, and Core Plate. The following components have been evaluated and determined to be structurally adequate to withstand the effects of flow induced vibrations: Guide Rods, Top Head Instrument Nozzle, Head Spray Nozzle, Top Head Vent Nozzle, Core Spray Sparger, Core Spray Piping, Fuel Assembly, and Shroud Head Bolts, Steam line Nozzle, Water Level Instrument Nozzle, and Top Guide.							
STP 35	Recirculation and Jet Pump System Calibration: A closely controlled pressure was applied simultaneously to an entire loop to obtain an integrated calibration check of the system instrumentation. Actual calibration of the jet pump flow instrumentation was completed during hot pressurized operation by comparison of the single and double tapped pressure drops	Y	Y	Y	Power Testing	Yes (Standard plant procedure)	The original testing purpose was to perform a complete calibration of the installed recirculation system flow instrumentation. The flow instrumentation was recalibrated when the new digital recirculation flow control system was installed. Since the total core flow does not change for EPU conditions, the recirculation jet pumps will not require recalibration for EPU conditions. If there are any indications or requiring jet pump recalibration, it would be performed using a standard plant	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	as a function of flow.						procedure.							
STP 36	Equalizer Open: Testing was performed to explore one pump operation with the equalizer valve open. Initial valve opening was made at a high pump speed by rapid jogging until the inactive loop jet pumps go from reverse to forward flow. Successive valve openings and pump speed increases were schedules to avoid pump loop P, pump speed and pump motor current limits. When the valve was full open or when limits are reached the available operating region was explored and data obtained. The test was concluded by rapidly closing the equalizer valve while recording the transient.	Y	N	N	Power Testing	No	The original testing purpose was to (a) explore the allowable operating range and performance of the recirculation system under conditions of one pump operation with the equalizer line valves open, and (b) to develop operating procedures for one pump equalizer open operation. The Unit 1 and 3 equalizer valves were removed during the recirculation piping replacement during the recovery. The Unit 2 valves, although still existent, are restricted from being opened during normal operation. Therefore, this test is no longer applicable and is not required for EPU.	NA						
STP 39 (Unit 1 only). See STP 9 for Units 2/3)	Water Level Verification in Reactor Vessel: The first part of testing measured the Yarway reference to verify agreement with the temperature correction factor used in calibration. The second part verified the ability of the feedwater control system	Y	N	N	Initial Heatup, Power Testing	Yes (Standard plant procedure)	The purpose of the original testing was to a) verify the calibration and agreement of the various narrow and wide range level indicators under various conditions and b) to demonstrate the ability of the feedwater control system to regulate reactor water level. As part of the restart efforts, the Yarway temperature equalizing columns are		X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	to regulate reactor water level at 50 percent flow/50 percent power and 100 percent flow/100 percent power.						removed and the instruments replaced with analog trip system devices. The hardware associated with this new reactor water level instrumentation is not modified and normal operational water level and level setpoints (alarms/trips/actuations) are not changed by EPU. The demonstration of procedures and operational validation for EPU therefore need not be repeated. Demonstration of the ability of the feedwater control system to regulate reactor water level is accomplished by standard plant procedures.							
STP 70	Reactor Water Cleanup System: Testing was performed to demonstrate the operability of the reactor water cleanup system under actual reactor operating temperature and pressure.	Y	Y	Y	Initial Heatup	Yes (Standard plant procedure)	The original testing purpose was to demonstrate the operability of the RWCU system under actual reactor operating temperature and pressure. As described in the EPU safety analysis report Sections 3.10, RWCU system operation at EPU slightly decreases the temperature within the system and the FW iron input to the reactor increases as a result of the increased FW flow. The effects of EPU on the system function have been reviewed and it was determined the system can perform adequately during EPU with the original RWCU flow. Therefore, no changes are required to meet the performance requirements for the system. Confirmation that the system meets TS	X	X	X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							requirements for operability for startup and power ascension is required by the refueling test program.							
STP 71	Residual Heat Removal System: Testing was performed to demonstrate the ability of the RHR system to remove residual and decay heat from the nuclear system so that refueling and nuclear system servicing can be performed, and remove heat from the pressure suppression pool water.	Y	Y	Y	Power Testing	Yes (Standard plant procedure)	The original testing purpose was to demonstrate the ability of the RHR system to (a) remove residual and decay heat from the nuclear system so that refueling and nuclear system servicing can be performed and (b) remove heat from the pressure suppression pool water. As described in the EPU safety analysis report Sections 3.9, the RHR system is designed to restore and maintain the reactor coolant inventory following a LOCA and remove decay heat following reactor shutdown for normal, transient, and accident conditions. The EPU effect on the system is a result of the higher decay heat in the core corresponding to the uprated power and the increased amount of heat discharged. The effects of the system functional capability have been reviewed and it was determined that the system functional basis continues to ensure that accident containment temperature limits are not exceeded and only plant availability is effected for normal reactor shutdown. Therefore, no changes are required to meet the performance requirements for the system. Confirmation that the system meets TS requirements for operability for	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							startup and power ascension is required by the Technical Specifications.							
STP 72	Drywell Atmosphere Cooling System: The Drywell Atmosphere Cooling System was placed in operation and its ability to maintain the temperature in the drywell.	Y	Y	Y	Power Testing	Yes (Standard plant procedure)	The original testing purpose was to verify the ability of the Drywell Atmosphere Cooling System to maintain design conditions in the drywell during operating conditions. As discussed in the EPU safety analysis report Section 6.4.3, the change in vessel temperature is minimal and does not result in any significant increase in drywell cooling loads. The testing performed during the normal refueling test program will meet the intent of the original test objectives during startup and power ascension at EPU.	X	X	X	X	X	X	X
STP 73	Cooling Water Systems: Testing was performed to verify that the performance of the RBCCW and raw cooling water systems is adequate with the reactor at rated condition.	Y	Y	Y	Initial Heatup, Power Testing	Yes (EPU startup test.)	The original testing purpose was to verify the performance of the RBCCW and the RCW systems is adequate with the reactor at rated condition. As described in the EPU safety analysis report Sections 6.4.3 and 6.4.4, the EPU heat load increases for these systems are minimal and sufficient heat load removal capacity is available with the current system flow rates to accommodate heat removal at EPU normal operating conditions. Therefore, no changes are required to meet the performance requirements for the system and system testing is governed by			X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGI-NAL TEST NUM-BER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
							approved plant procedures which remain valid for EPU operations.							
STP 74	Off Gas System: Testing was performed to verify the proper operation of the Off Gas system over its expected operating parameters and to determine the performance of the activated carbon adsorbers.	N	Y	Y	Power Testing	Yes (Standard plant procedure)	The original testing purpose was to verify the proper operation of the Off Gas system over its expected operating parameters and to determine the performance of the activated carbon adsorbers. As described in the EPU safety analysis report Section 8.2.1, the EPU hydrogen flow rates and concentrations are still within the design limits of the system and the system components have sufficient design margin to handle the increase in thermal power for EPU with exceeding the system design limits of temperature, flow rates, or heat loads. Therefore, no changes are required to meet the performance requirements for the system. Confirmation that the system meets requirements for operability for startup and power ascension is required by a standard plant procedure.	X	X	X	X	X	X	X
STP 75	Reactor Shutdown from Outside the Main Control Room: With the plant operating at greater than 10 percent generator output, the reactor was scrammed by closing the MSIVs from the backup control	N	N	Y	Power testing	Yes (Standard plant procedure)	The purpose of this test was to demonstrate that the plant was designed and constructed with adequate instruments and controls to permit safe reactor shutdown from outside the main control room and maintain it in a safe condition, that the minimum number of personnel required by the TS are adequate without affecting the safe,	X						

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	station. Operators manned their backup control stations as described in the emergency operating instructions. The RCIC system was operated from the backup controls to supply water to the reactor vessel. Suppression pool cooling was placed in operation using the backup controls. An extra licensed operator remained in the main control room to assure that the test was terminated and control returned to normal in the event of any unexpected conditions occurred. The test was terminated when it was assured that the reactor could be maintained in a safe hot standby condition from the backup controls.						continuous operation of the other units, and that the plant emergency operating instructions are adequate. As described in the EPU safety analysis report Section 10.6, EPU does not change any of the plant automatic safety functions. Also, after the applicable automatic responses have initiated, the subsequent operator actions for plant safety do not change for EPU. Additional operator training will be required to enable plant operation at the EPU power level. Training required to operate the plant following EPU will be conducted prior to restart of the units at the EPU conditions. Standard plant procedures that are performed once per operating cycle ensure that if the control room becomes inaccessible, the plant can be placed and maintained in MODE 3 from the backup control panel and the local control stations.							
STP 92	Steam Separator-Dryer: Samples were taken from the inlet and outlet of the steam dryers, and the inlet at the steamline at various power levels at chosen water levels and recirculation flow rates. The amount of carryunder was estimated from these samples and carryover was determined	Y	N	N	Open Vessel Testing, Initial Heatup, Power Testing	Yes (EPU startup test)	The original startup testing took samples from within the vessel and the inlet at the steam line for the determination of carryunder and carryover. EPU testing will determine steam separator-dryer moisture carryover. For this testing, MSL moisture content is considered equivalent to the steam separator-dryer moisture carryover. This MSL moisture content test data will determine the steam separator-dryer performance (i.e.,			X	X	X	X	X

TABLE 1
COMPARISON OF BFN INITIAL STARTUP TESTING AND PLANNED EPU TESTING

ORIGINAL TEST NUMBER	ORIGINAL TEST DESCRIPTION (DERIVED FROM UFSAR SECTION 13.5)	ORIGINAL TEST UNIT			ORIGINAL S/U TEST PHASE	TESTING PLANNED FOR EPU	EVALUATION / JUSTIFICATION / NOTES[1]	EPU TEST CONDITIONS PERCENT OF 3293 MWT (OLTP)						
		U1	U2	U3				<90	90[2]	100[3]	105	110	115	EPU
	from Na-24 activities in samples taken from the outlet of the steam dryers.						moisture carryover) for the EPU and core conditions in effect at the time of the test. This testing will meet the intent of the original test objectives.							

Notes:

1. In this table, “EPU Safety Analysis Report” refers to the PUSAR for Unit 1 and the PUSAR and/or the FUSAR for Units 2 and 3 as applicable.
2. Power relative to Current Licensed Thermal Power (CLTP).
3. Column applicable to Unit 1 only.

TABLE 2
COMPARISON OF SRP 14.2.1 TEST MATRIX AND BFN INITIAL STARTUP TESTS

SRP Power Ascension Test	BFN Initial Startup Test? (Test Numbers)
Steady-State Power Ascension Testing	
Conduct vibration testing and monitoring of reactor vessel internals and reactor coolant system components	Yes (34, 90)
Measure power reactivity coefficients (PWR) or power vs. flow characteristics (BWR)	Yes (30)
Steady-state core performance	Yes (19)
Control rod patterns exchange	No (Performed Analytically)
Control rod misalignment testing	N/A (PWR only)
Failed fuel detection system	No (No Applicable System)
Plant process computer	Yes (13)
Calibrate major or principal plant control systems	Yes (14, 15, 22, 23, 24, 29, 32)
Main steam and main feedwater system operation	Yes (22, 23)
Shield and penetration cooling systems	No (Standard Procedure)
ESF auxiliary and environmental systems	No (Standard Procedure)
Calibrate systems used to determine reactor thermal power	No (Standard Procedure)

TABLE 2
COMPARISON OF SRP 14.2.1 TEST MATRIX AND BFN INITIAL STARTUP TESTS

SRP Power Ascension Test	BFN Initial Startup Test? (Test Numbers)
Chemical and radiochemical control systems	Yes (1)
Sample reactor coolant system and secondary coolant systems	Yes (1)
Radiation surveys	Yes (2)
Ventilation systems (including primary containment and steam line tunnel)	Yes (72)
Acceptability of reactor internals, piping, and component movement, vibrations, and expansions	Yes (17, 34, 90)
Transient Testing	
Relief valve testing	Yes (26)
Dynamic response of plant to design load swings	Yes (22, 23, 25, 27, 29, 30, 32)
Reactor core isolation cooling functional test	Yes (14)
Dynamic response of plant to limiting reactor coolant pump trips or closure of reactor coolant system flow control valves (Reactor coolant recirculation pump trip test)	Yes (30)
Dynamic response of the plant to loss of feedwater heaters that results in most severe feedwater temperature reduction	No (Performed Analytically)
Dynamic response of plant to loss of feedwater flow	Yes (23)
Dynamic response of plant for full load rejection (Loss of Offsite Power Testing)	Yes (27, 31)

TABLE 2
COMPARISON OF SRP 14.2.1 TEST MATRIX AND BFN INITIAL STARTUP TESTS

SRP Power Ascension Test	**BFN Initial Startup Test?** (Test Numbers)
Dynamic response of plant to turbine trip (Turbine trip or generator trip)	Yes (27, 31)
Dynamic response of plant to automatic closure of all main steam isolation valves	Yes (25)

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
Main Turbine	• Replace HP Turbine diaphragms and rotor buckets • Replace HP Rotor/LP Rotors (Unit 1 only) • Replace springs, bonnets, washers, bellows, & bolting on 6 cross around relief valves to permit increased set pressure • Replace miter bend elbows in the condenser spray piping with long radius elbows to reduce back pressure	No	No	No	• Turbine Balancing (if required) • Overspeed Test • Control and Stop Valve testing • Relief valve bench testing
Turbine Sealing Steam	• Modify the size of the steam seal unloader valves and associated piping to allow the turbine sealing system to accommodate the larger steam flow requirements	No	No	No	• Condenser Vacuum testing monitor steam seal header pressure • Calibration of the Steam Seal Header Pressure controller • Inservice leak test

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
Condensate Pumps	• Replace 2 impellers in each of 3 pumps • Install 3 - 1250 hp motors • Recalibrate relay settings • Recalibrate/replace pump & motor instrumentation • Modify HVAC ductwork	Yes	Yes	Yes	• Verification of pump flow and head • Monitoring of pump and motor parameters (flow pressure, temperatures, etc.) • Instrumentation calibration and functional testing
Condensate Booster Pumps	• Replace 3 pumps • Install 3 – 3000 hp motors • Recalibrate relay settings • Recalibrate/replace pump & motor instrumentation • Modify HVAC ductwork	Yes	Yes	Yes	• Verification of pump flow and head • Monitoring of pump and motor parameters (flow pressure, temperatures, etc.) • Instrumentation calibration and functional testing

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
Feedwater Pumps and Turbines	• Replace 3 pumps • Recalibrate pump instrumentation and control system for increased flows at EPU conditions • Replace turbine/pump coupling • Replace turbine rotor, diaphragms and buckets • Recalibrate/replace turbine instrumentation	Yes	Yes	Yes	• Balancing • Overspeed testing Controls Tuning • Verification of pump flow and head • Monitoring of pump and turbine parameters (flow pressure, temperatures, etc.) • Instrumentation calibration and functional testing
Moisture Separators	• Change vanes and add perforated plate on moisture separators • Modify internal drains as needed	No	No	No	• Moisture removal effectiveness testing • Inservice leak test • Installation testing (flow, temperature pressure, etc.)

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
Feedwater Heaters	• Upgrade heater shell pressure certification • Rerate tube side pressure certification for FWH 3 • Replace level transmitters on FWHs 1, 2 & 3 • Repair / replace 18 nozzles on FWHs 1, 2 & 3 • Replace relief valves on FWHs 1, 2 & 3 • Relocate extraction steam nozzle & shorten extraction steam line on FWH 3 • Install new impingement plate & steam duct inside FWH 3 • Reinforce / re-weld pass partition plates in all FWHs • Install manway stiffeners on FWH 3	No	No	No	• Relief Valve bench testing • Installation Testing (flow, temperature, pressure, etc.) • Instrumentation calibration and functional testing • Inservice leak rest.
Main Condenser Extraction Steam Bellows	• Replace #2, #3 and #4 bellows with upgraded bellows	No	No	No	• Installation Examination

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
Condensate Demineralizers	• Install 1 new vessel with valves & digital controls • Upgrade controls on 9 existing vessels to digital (Unit 2 only) • Install digital control on 9 existing vessels (Unit 1 only) • Replace valves for increased reliability	No	No	No	• Control system functional testing • Initial installation Startup test (flow, temperature, pressure, etc.)
Steam Packing Exhauster Bypass	• Install 24” piping & flow control valve to accommodate increased condensate flows at EPU conditions	No	No	No	• Valve testing
Drywell Building Steel	• Modify building steel beams and connections as required for load changes at EPU conditions	Yes	Yes	No	• Applicable structural installation testing
Torus Attached Piping	• Modify supports and snubbers as required due to EPU conditions	Yes	Yes	No	• Applicable structural installation testing
Main Steam Supports	• Modify supports as required for load changes due to EPU conditions	Yes	Yes	No	• Applicable structural installation testing • Vibration monitoring of Main Steam and Feedwater Piping and components

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
Steam Dryer	• Perform identified modifications required to maintain Dryer structural integrity at EPU Conditions	No	No	No	• Testing is discussed in TVA's February 23, 2005 letter, Reply 5.a (1)
Main Steam Relief Valves (Unit 1 only)	• Increase mechanical setpoint 30 psi with 3% tolerance due to increased reactor pressure	Yes	Yes	No	• Bench testing of setpoints • Remote manual opening at operating reactor pressure
Motor Operated Valves (Unit 1 only)	• Modification to GL 89-10 MOVs to accommodate 30 psi pressure increase	Yes	Yes	No	• MOVATS
Reactor Recirculation Pump Motors	• Revise electrical protection system setpoints • Revise temperature monitoring setpoints • Assess additional heat load on plant HVAC & cooling water systems • Assess power cable voltage drop increase due to higher current • Revise pump/motor vibration monitoring setpoints • Re-rate pumps and motors for 120% Power/105% Core Flow operating conditions	Yes	Yes	No	• Applicable instrumentation calibrations • Vibration monitoring • Controls tuning and system operation during vessel hydro

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
Jet Pumps	• Install sensing line clamps as required to reduce vibration due to Recirculation Pump vane passing frequencies at EPU conditions	Yes	Yes	No	• None required
Main Generator System	• Recalibrate/replace pressure regulators and pressure switches • Increase generator hydrogen to 75 psig to operate at increased loads • Rewind generator stator and generator field (Unit 1 only)	No	No	No	• Field Installation testing • Instrumentation calibration and functional testing • Monitoring of system (i.e., voltage, amps, temperature) during power ascension
Isolation Phase Bus Duct Cooling	• Modify Isolation Phase Bus Duct Cooling System to remove Bus Duct heat under EPU conditions	No	No	No	• Verification of system flow, both air and water
Main Bank Transformers	• Install 3-500 MVA transformers per unit • Install 2-500 MVA spares (U1/2 & 3) • Upgrade oil and water deluge systems • Upgrade relaying as needed • Replace 5 – breakers & disconnects	No	No	No	• Field Installation testing • Deluge spray down testing • Performance monitoring
ICS/SPDS	• Update as needed based on NSSS and BOP instrument changes	No	No	No	• Performance monitoring

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
Reactor Recirculation Vibration Monitoring (Unit 2)	• Install ~ 35 temporary sensors on RR, RHR, & RWCU in Drywell. • Install data acquisition center in Rx Bldg. • Conduct testing program before EPU outage. (Installed on Unit 2 and vibration data obtained during startup from U2C12 Outage.)	No	No	No	• Collect and analyze vibration data
Vibration Monitoring	• Install temporary sensors based on ongoing analyses • Conduct testing program during power ascension	No	No	No	• Collect and analyze vibration data on selected systems
Main Steam Isolation Valves	• Replace MSIV poppets and modify operators (Unit 1 only) as required to reduce differential pressure across MSIVs at EPU conditions • Install 2-inch MSIV stems as required due to increased stem forces caused by EPU MS flow increase	Yes	Yes	Yes	• Stroke time testing • Applicable Technical Specifications testing • Performance monitoring
EHC Software	• New program inputs & logic for EPU conditions	Yes	Yes	Yes	• Verification of control functions • Turbine Valve setup • Controls Tuning

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
High reactor vessel pressure scram setpoint (Unit 1 only)	• Raise setpoint due to increased normal reactor vessel pressure Technical Specification value	Yes	Yes	No	• Calibration per applicable Surveillance Requirements instructions
High reactor vessel pressure trip for the reactor recirculation pump (Unit 1 only)	• Raise setpoint due to increased normal reactor vessel pressure Technical Specification value	Yes	Yes	No	• Calibration per applicable Surveillance Requirements instruction
ATWS-RPT reactor vessel pressure recirculation pump trip setpoint (Unit 1 only)	• Raise setpoint due to increased normal reactor vessel pressure Technical Specification value	Yes	Yes	No	• Calibration per applicable Surveillance Requirements instructions
Standby Liquid Control System pump discharge test pressure (Unit 1 only)	• Raise discharge pressure requirement due to increased normal reactor vessel pressure	Yes	Yes	No	• Verified by Surveillance Requirement testing
Turbine Stop Valve – Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure – Low bypass setpoint	• Lower percent reactor thermal power to maintain trip at same absolute thermal power	Yes	Yes	No	• Calibration per applicable Surveillance Requirements instructions
Neutron flux – High (set down)	• Lower percent reactor thermal power to maintain trip at same absolute thermal power	Yes	Yes	No	• Calibration per applicable Surveillance Requirements instructions

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
Flow Biased Simulated Thermal Power - High	• Adjust slope and intercept for equation to reflect change in maximum reactor thermal power	Yes	Yes	No	• Calibration per applicable Surveillance Requirements instructions
RCIC and HPCI surveillance testing pump discharge pressure (Unit 1 only)	• Raise discharge pressure requirement due to increased normal reactor vessel pressure	Yes	Yes	No	• Verified by Surveillance Requirement testing
Main Steam Line high flow	• Adjust delta P to account for higher flow rate.	Yes	Yes	No	• Calibration per applicable Surveillance Requirements instructions
ATWS Alternate Rod Injection (ARI) high reactor pressure setpoint (Unit 1 only)	• Raise setpoint due to increased normal reactor vessel pressure Technical Specification value	Yes	Yes	No	• Calibration per applicable Surveillance Requirements instructions
Steam/Feedwater Normal Flow Rate	• Increased flow rate to accommodate increased reactor thermal power output	Yes	No	Yes	• Monitored to ensure plant remains within anticipated operational limits
Feedwater and Reactor Vessel Temperature Changes	• Feedwater temperature increase • Reactor vessel temperature decrease due to increased feedwater flow rate	Yes	No	No	• Change in temperature accounted for in instrument calibration.

Table 3
Browns Ferry EPU Planned Modifications, Setpoint Adjustments and Parameter Changes

Activity	Description	Impacts Function Important to Safety	Required to Mitigate Plant Transient	Response Action Involves Multiple SSCs	Testing
Recirculation Pump Flow Rate	• Increased required recirculation pump flow rate required to achieve total core flow	No	No	No	• Verification of total core flow
Reactor Pressure (Unit 1 only)	• 30 psi increase in normal operating steam dome pressure	Yes	No	No	• Main Steam Relief Valves Bench test • Generic Letter 89-10 valves MOVATS tested • Reactor Vessel pressure test • HPCI/RCIC flow tests • Standby Liquid Control System test pressure increased
Primary Containment Pressure and Torus Temperature Post Accident	• Peak pressure and temperature change	See Testing Column	See Testing Column	See Testing Column	• Analytically demonstrated to achieve successful system operation (i.e., primary containment, ECCS, etc) since actual post accident conditions cannot be simulated • Integrated containment Leak rate testing • Local leak rate tests • ECCS Flow tests