Attachment 46

Startup Test Plan

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1.0 Introduction

This document provides information to supplement the Browns Ferry Nuclear Plant (BFN) EPU License Amendment Request (LAR), and provides additional information about startup testing using SRP 14.2.1- Generic Guidelines for Extended Power Uprate Testing Programs as a guide.

2.0 Purpose

2.1. Background

BFN Units 1, 2, and 3 are General Electric boiling-water reactors (BWR/4) with Mark-I containments. All three units were originally licensed for operation at 3293 MWt. Units 1, 2, and 3 commenced operation in 1973, 1974, and 1976, respectively. The units shut down in 1985 to address management, technical, and regulatory issues. Units 2 and 3 restarted in 1991 and 1995, respectively, and have been in operation since then, Units 2 and 3 were authorized to increase their maximum power by 5-percent (to 3458 MWt) in 1998. Unit 1 remained shut down between 1985 and 2007. In 2007, Unit 1 restarted, and TVA received authorization to increase BFN Unit 1 maximum power by 5-percent (to 3458 MWt). Unit 1 has been in operation since 2007.

This document provides detailed information on the testing TVA intends to perform following the extended power uprate (EPU) implementation outages at BFN Units 1, 2, and 3. BFN plans to implement a Constant Pressure Power Uprate (CPPU) to 3952 MWt. The planned uprate is approximately fourteen percent (14%) above current licensed thermal power (CLTP) and twenty percent (20%) above the original licensed thermal power (OLTP). The planned EPU implementation outages are in spring 2018, fall 2018, and spring 2019 for BFN Units 3, 1, and 2 respectively. Following each of these outages, TVA will conduct a comprehensive EPU startup test program to ensure the safe operation of each plant. This attachment describes that test program.

Modifications related to EPU may be installed before or during the above referenced outages. Additional modifications, unrelated to EPU, may also be installed during these outages and require testing during startup. A power ascension test procedure will be developed that incorporates the tests from this EPU startup test plan as well as any testing required from other modifications installed during these outages. This includes modifications that support EPU's increased power generation in the nuclear boiler system and secondary plant with higher performance requirements due to increases in flow rates, heat loads, electrical loads and power production, and core power, as applicable.

TVA, with this test plan, plant procedures, and processes, will assure the functions of plant equipment important to safety are adequately demonstrated prior to operation at the EPU power level.

The required EPU startup testing for BFN was developed using information from several sources:

- The startup and power ascension testing performed during the initial plant startup as discussed in the BFN UFSAR Section 13.5.
- Testing performed following each refueling outage as discussed in the BFN UFSAR Section 13.10.
- The guidance for extended power uprates in GE topical report NEDC-32424P-A "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (also referred to as ELTR1), including the NRC's Requests for Additional Information, GE responses and the NRC's staff position on the ELTR1 documented in NRC letter dated February 8, 1996.
- The guidance for extended power uprates in GE topical report NEDC-32523P-A "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (also referred to as ELTR2), including Supplement 1, Volumes 1 and 2 and the NRC's staff position on the ELTR2 documented in NRC letter dated September 14, 1998.
- The guidance for constant pressure power uprates (CPPU) in GE Topical Report NEDC-33004P-A "Constant Pressure Power Uprate" (also referred to as CLTR), including the NRC's Requests for Additional Information, GE responses and the NRC's safety evaluation of the CLTR documented in NRC letter dated March 31, 2003.
- The NRC Standard Review Plan (SRP), NUREG-0800, Section 14.2.1 "Generic Guidelines for Extended Power Uprate Testing Programs."
- Information and data from plant transients that occurred during BFN's operating history as applicable.
- Experience from other BWR plants that have implemented EPUs.

BFN Units 1, 2, and 3 have a similar system geometry, reactor protection system, configuration and mitigation features. Additionally, BFN Units 1, 2, and 3 have similar thermal-hydraulic and transient behavior characteristics. Therefore, test methods and acceptance criteria will be the same for all units.

The NRC endorsed General Electric Licensing Topical Reports (ELTR1 and ELTR2) for Extended Power Uprates. The NRC also accepted the test program of the CPPU Licensing Topical Report (CLTR) for EPUs, but reserved the right to consider on a plant specific basis the CLTR recommendations not to perform Large Transient Testing. The CLTR is the controlling document for the BFN's planned EPU. For BFN, TVA will comply with the startup test requirements of the CLTR, and will take exception to performing Large Transient Testing as discussed in ELTR1. Accordingly, this document addresses eliminating large transient testing for BFN Units 1, 2, and 3.

2.2. Objective

This attachment describes the startup testing TVA will conduct associated with implementation of EPU at BFN, including justification for not performing large transient testing. The presentation of information in this attachment is consistent with SRP Section 14.2.1.

3.0 Summary of Conclusions

Based on the discussions in Section 5, Table 46-1, and Table 46-2 of this attachment, BFN will conduct an EPU test program that provides a controlled ascension to the proposed EPU power level. The test program will include sufficient testing to demonstrate that structures, systems, and components will perform satisfactorily at EPU power levels. The EPU startup test plan is based on the initial startup test program. Tests described in UFSAR Section 13.5 were reviewed, and no tests were identified as being invalidated by EPU. The EPU startup test plan is integrated with other modification testing to demonstrate that all modifications have been adequately implemented.

BFN will develop the post modification testing for each of the modifications, listed in Attachment 47 to this LAR, in accordance with the BFN modification program. Performance testing for these, and any other modifications installed during the EPU outages, will be integrated into a single, controlling Power Ascension Test Procedure, as necessary, to ensure the aggregate effect of EPU and all modifications do not affect the safety performance of BFN Units 1, 2, and 3.

<u>Initial Plant Startup Testing and Comparison to EPU Testing</u>

Table 46-1 to this enclosure shows the initial plant startup testing performed at Browns Ferry, and compares it to EPU startup testing. Per the guidance of SRP Section 14.2.1, Table 46-1 to this attachment identifies all of the tests that were performed during initial startup and the approximate power levels at which the tests were performed. There were no tests at lower power levels that would be invalidated by EPU. For each initial startup test at power ≥ 80 percent Original Licensed Thermal Power (OLTP), Table 46-1 indicates how that test is addressed either with testing or provides the justification for not performing that test.

Large Transient Testing for BFN Units 1, 2, and 3, (See Section 5) is not required for EPU because:

1) BFN previously performed Large Transient Tests, as part of the initial startup test program, and documented the results;
2) Potential gains from further Large Transient Testing are minimal and produce an unnecessary and undesirable transient cycle on the primary system;
3) Analytical methods and training facilities adequately simulate large transient events without the need to impose actual events;
4) Plant operators will be trained in potential EPU transient events through the use of simulator models containing Balance of Plant (BOP) transients;
5) Probabilistic Risk Assessment (PRA) analysis indicates an increased risk of core damage and large early release if the tests are performed; and

6) Industry operating experiences indicate that plants will continue to respond to these transients as designed following EPU implementation.

TVA's decision to exclude Large Transient Testing from BFN's EPU Test Plan is consistent with the CLTR and most BWR EPU test plans.

In view of previous test results and the plant response to prior documented events, the EPU startup testing program, as proposed in this attachment, is considered acceptable to validate the continued ability of the plant to safely operate within required parameters and operational limits.

TVA requests the NRC approve the exception to perform Large Transient Testing for BFN's EPU. TVA has concluded that BFN and industry data provide an adequate correlation to allow the effects of the EPU to be analytically determined on a plant specific basis.

Tests Associated with EPU Plant Modifications

Modifications associated with EPU are described in Attachment 47 of the EPU LAR. Testing related to changes will be performed in accordance with the plant's modification program. The intent of the testing associated with each modification is to demonstrate that structures, systems, and components (SSCs) affected by the modification conform with the as-designed requirements of modifications and to ensure overall system integrity is not adversely affected by changes in design.

As discussed in Section 2.1, EPU modifications may be installed and in-service before the scheduled EPU implementation outages. As shown in Attachment 47 to this LAR, most modifications either are installed or will be installed prior to the actual EPU implementation outages. As result, the significant power block modifications (feedwater pumps, condensate pumps, condensate booster pumps, condensate demineralizers, etc.) will have several years of operating and maintenance experience prior to EPU. Therefore, it is anticipated that modification testing during the initial power ascension to full EPU power will be minimal.

The use of a constant pressure power uprate (CPPU) approach minimizes the changes in operating conditions plant systems will experience with EPU. The guidance in the CLTR was utilized to determine the extent of testing associated with the EPU modifications.

EPU Testing

Table 46-2 summarizes the planned EPU power ascension tests.

4.0 Testing Evaluations

The CLTR provides the following guidance: 1) The same performance criteria will be used for EPU as in the original power ascension tests unless they have been replaced by updated criteria since the initial test program; and 2) because reactor operating pressure and current licensed maximum core flow do not change, testing of system performance affected by steam pressure or core flow is not necessary with the exception of the tests listed in Section 10.4 of the CLTR. The testing planned for BFN to support implementation of EPU conforms to the guidance provided in the CLTR.

4.1. Comparison to BFN Original Startup Test Program (SRP 14.2.1:III.A)

Table 46-1, "Comparison of BFN Initial Startup Testing and Planned EPU Testing", identifies all startup tests performed during initial startup of BFN Units 1, 2, and 3. These tests include all those performed with the vessel head off, during heat-up, and during power ascension. The tests were divided into two groups: tests performed below 80 percent power, and tests performed at 80 percent of OLTP or greater.

Tests performed below 80 percent OLTP were reviewed to ensure none of those tests would be invalidated by the EPU. Tests performed at 80% of OLTP or greater were evaluated to determine a) the original tests are not invalidated, and b) EPU testing adequately addresses these tests.

<u>Power ascension tests performed at < 80% of OLTP</u>

In accordance with SRP Section 14.2.1, paragraph III.A.2, the initial startup tests conducted at power levels below 80 percent OLTP were reviewed for tests that would potentially be invalidated by EPU. This review was accomplished by examining the test descriptions in UFSAR Section 13.5, the test conditions provided in UFSAR Tables 13.5-1 through 13.5-6, and the startup test reports for all three BFN units, and then compare the initial tests against plant changes due to EPU. At low power, no tests are invalidated by the BFN EPU since operation at low power has not changed for EPU. Therefore, it is not necessary to include these tests in the EPU startup test plan. As shown in Table 46-1, and referring to the test numbers in UFSAR Sections 13.5.2 and 13.5.3, the following tests were performed exclusively below 80 percent OLTP: No.3, No.4, No.6, No.10, No.14, No.15, No.16, No.17, No.26, No.31, No.70, No.71, and No.75.

<u>Power ascension tests performed at ≥ 80% of OLTP</u>

Table 46-1, Comparison of BFN Initial Startup Testing and Planned EPU Testing, provides a comparison of the initial startup tests to the planned testing for the EPU. As shown in Table 46-1, and referring to the test numbers in UFSAR Sections 13.5.2 and 13.5.3, the following tests were performed at 80% of OLTP or greater: No. 1, No. 2, No. 5, No. 9, No. 11, No. 12, No. 13, No. 18, No. 19, No. 20, No. 21, No. 22, No. 23, No. 24, No. 25, No. 27, No. 29, No. 30, No. 32, No. 33, No. 34, No. 35, No. 36, No. 39, No. 72, No. 73, No. 74, No. 90, and No. 92.

Additional details for planned EPU testing are provided in Table 46-2, Planned EPU Power Ascension Testing. Justifications for not performing certain testing are provided in Section 5 of this attachment. A listing of transient tests performed at 80% or greater during initial startup testing is provided below.

<u>Power ascension transient tests performed at ≥ 80% of OLTP</u>

Table 4-1 shows startup transient tests performed at 80% OLTP or greater. This table is provided in accordance with SRP Section 14.2.1, paragraph III.A.1 and III.A.2. Initial startup tests, along with test power levels, are also provided in Table 46-1 of this attachment.

Table 4-1 Initial Startup Transient Tests Performed at ≥ 80% OLTP

Test Number	Initial Startup Transient Test	Max Power Level (% of OLTP) U1	U2	U3	Table 1 or 2 of SRP Section 14.2.1	EPU Testing Planned
22	Pressure Regulator					
	• Step Setpoint Changes	100	100	100	Table 1	Yes
23	Feedwater System					
	• Single Pump Trip	100	100	100	Table 2	No*
	• Step Setpoint Changes	100	100	100	Table 1	Yes
	• Loss of FW Heating	85	----	----	Table 2	No*
24	Bypass Valves					
	• Cycle Turbine Bypass Valves	100	100	100	Table 1	Yes
25	Main Steam Isolation Valves					
	• Closure of Single MSIV	70	70	70	Table 1	No*
	• Closure of All MSIVs	70	100	100	Table 2	No*
27	[Unit 1] Turbine Stop and Control Valve Trips					
	• Turbine Trip	100	100	----	Table 2	No*
	• Control Valve Trip	80	100	----	Table 2	No*
	[Units 2 and 3] Turbine Trip and Generator Load Rejection					
	• Turbine Trip	----	----	75	Table 2	No*
	• Generator Trip	----	----	100	Table 2	No*
30	Recirculation System					
	• 1 Pump and 2 Pump Trips	100	100	100	Table 2	No*

* See Section 5 for justification.

As shown in Table 4-1 above and Table 46-1 to this attachment, Test No. 22, Test No. 23 (step setpoint changes only), and Test No. 24 (Cycle Turbine Bypass Valves) will be included as part of EPU startup testing. Test No. 23 (Feedwater Pump Trip and Loss of FW Heating), Test No. 25 (MSIV Closures), Test No. 27 (Turbine and Generator Trips), and Test No. 30 (Recirculation Pump Trips) will not be tested. Section 5 of this attachment provides the justification for eliminating these transient tests.

4.2. Post Modification Testing Requirements (SRP14.2.1:III.B)

Attachment 47 of the EPU LAR provides a listing of EPU implementation modifications. BFN plans to complete the necessary modifications to achieve 120% OLTP prior to the conclusion of the spring 2018 outage for BFN Unit 3, the fall 2018 outage for BFN Unit 1, and the spring 2019 for BFN Unit 2. An EPU startup test program will be conducted following each unit's outage.

Modification Aggregate Impact

As can be seen from an inspection of the modifications listed in Attachment 47 of the EPU LAR, most of the modifications are set point changes to maintain comparable differences between system setting and actual limits, or typical EPU component replacements to accommodate the increased flows associated with an EPU. These modifications do not change system function, and installation maintains design margin at EPU conditions.

The High Pressure Turbine Replacement Modification, the Condensate Pump Modifications, the Condensate Booster Pump Modifications, and the Feedwater Pump Modifications have an impact on the reactor plant as they are directly tied to the primary system piping, and steam flow to the turbine is increased by approximately 16%. However, their function and interrelationship is essentially unchanged.

Condensate and Feedwater System upgrades represent significant plant modifications. These changes include the replacement of condensate pump internals and motors, the addition of a new filter demineralizer vessel, the replacement of Condensate Booster Pumps and motors, the replacement of Feedwater Pumps. Post modification testing will address individual changes, and Feedwater System Power Ascension Testing addresses the aggregate impact. See Table 46-2, Planned EPU Power Ascension Testing, for a description of planned Feedwater system testing.

Other modifications implemented at BFN as part of the EPU Project may not directly involve power generation. The details of associated post modification and startup testing will be developed in accordance with the BFN modification program. The performance attributes of these modifications, if any, will be verified during post modification testing, surveillance testing, and plant startup and operational testing, as applicable.

The startup test plan will be integrated into a power ascension test procedure that will include EPU tests, all other modification tests required during power ascension, and normal beginning of cycle refueling tests. The sequence of completing modifications and performing post-modification tests and test activities will coincide with the appropriate plant mode and power level and comply with all requirements of the BFN operating license to ensure a smooth orderly return to power and power escalation through completion of power ascension testing. Power ascension testing will include appropriate hold points to provide time to assess the plant response, verify test acceptance criteria compliance, and verify the test results and plant's operating performance at power levels above CLTP.

4.3. Startup Test Plan

The aggregate impact of EPU plant modifications, setpoint adjustments and parameter changes will be demonstrated by a test program established for a Boiling Water Reactor (BWR) EPU in accordance with startup test specifications as described in PUSAR Section 2.12.1, Approach to EPU Power level and Test Plan. The startup test specifications are based upon analyses and GE BWR experience with uprated plants to establish a standard set of tests for initial power ascension for EPU. These tests, which supplement the normal Technical Specification testing requirements and balance of plant monitoring, are summarized below:

- Testing will be performed in accordance with the Technical Specifications Surveillance Requirements on instruments re-calibrated for EPU conditions.
- Overlap between the IRM and APRM will be assured
- Testing will be done to confirm the power level near the turbine first stage scram bypass setpoint.
- EPU power increases will be made along established flow control/ rod lines in predetermined increments of ≤ 5% power starting from 90% up to 100 % of CLTP so system parameters can be projected for CPPU power before the CLTP is exceeded.
- Operating data, including fuel thermal margin, will be taken and evaluated at each step. Routine measurements of reactor and system pressures, flows, and vibration will be evaluated at each measurement point, prior to the next power increment.
- Radiation measurements will be made at selected power levels to ensure the protection of personnel.
- Control system tests will be performed for the reactor feedwater/reactor level controls and pressure controls. These operational tests will be made at the appropriate plant conditions for that test at each of the power increments, to show acceptable adjustments and operational capability.
- Steam dryer/separator performance will be confirmed within limits by determination of steam moisture content and by evaluating steam dryer dynamic loading during power ascension testing. The Steam Dryer Monitoring Program is discussed in Attachment 40 [proprietary] and Attachment 41 [non-proprietary].
- Vibration monitoring of main steam, feedwater and other balance of plant piping and components will be performed to permit a thorough assessment of the effect of EPU on the plant. The vibration monitoring program is discussed in Attachment 45.

<u>Conduct of Testing</u>

For Browns Ferry EPU, Operations is responsible for the implementing the Startup Test Plan. A Senior Reactor Operator will be designated as the EPU Startup Test Coordinator. Testing will be executed using plant approved procedures. As a minimum, test procedures will include prerequisites, instructions, and acceptance criteria. Some tests will be classified as Complex, Infrequently Performed Tests or Evolutions (CIPTE) in accordance with plant administrative procedures. For CIPTEs, plant site senior management will provide additional oversight during applicable tests. Management review and approval of test results at each power level will be provided prior to increasing power to the next level.

The same performance criteria will be used as in the original power ascension tests, except where they have been replaced by updated criteria since the initial test program. Specific test acceptance criteria may have changed from initial startup due to implementation of modifications. The revised acceptance criteria will be incorporated in the power ascension test procedure. Because dome pressure and core flow have not changed and recirculation drive flow will only increase slightly for EPU to achieve rated flow conditions, testing of system performance affected by these parameters is not necessary with the exception of the tests listed above.

The tests to be performed and the power levels at which they will be performed are described in Table 46-1 and Table 46-2. The overall power ascension test procedure is designed to provide management oversight and control of the testing activities to assure BFN can operate safely up to the licensed EPU thermal power level. EPU testing beyond CLTP is performed along an established flow control/rod line to ascend to EPU power in uniform increments of ≤ 5%. This incremental testing approach ensures a careful, monitored ascension to 100% EPU power. As power is raised to each plateau, tests are performed to demonstrate acceptable performance and power-dependent parameters are evaluated for acceptability. If all test results are satisfactory, the results will be assembled and presented to the BFN Plant Operations Review Committee (PORC) for approval prior to increasing power to the next level. The first review by the BFN PORC is required to occur prior to exceeding 100% CLTP with subsequent reviews completed prior to exceeding the next incremental step in power with a final review after reaching 100% EPU power.

EPU tests will have Level 1 and Level 2 acceptance criteria.

Level 1 criteria are associated with design performance. If a Level 1 test criterion is not met, then the plant will be placed in a hold condition that is judged to be satisfactory and safe, based upon prior testing. Tests consistent with this hold condition may be continued as permitted by Technical Specifications, operating procedures, and test procedures. Resolution of the problem will be immediately pursued by equipment adjustments or through engineering evaluation as appropriate. An evaluation will be initiated to identify the issue, document the cause of the issue, and obtain the actions necessary to correct the problem. (The process will be in accordance with the plant's Corrective Action Program.) The problem resolution plan will be presented to the BFN PORC for approval prior to implementing corrective actions. Following resolution, the applicable test portion will be repeated to verify the Level 1 criterion is satisfied. A description of the problem, resolution, and successful test will be included in a report documenting the issue. The report will be presented to the BFN PORC for approval prior to increasing reactor power.

Level 2 criteria are associated with performance expectations. If a Level 2 criterion is not met, then the plant operating condition or test plans would not necessarily be altered. An evaluation will be initiated to identify the issue, document the cause of the issue, and obtain the actions necessary to correct the problem. (The process will be in accordance with the plant's Corrective Action Program.) If equipment adjustments are required, the applicable test portion will be repeated to verify that the Level 2 criterion is satisfied. A description of the problem, resolution, and successful test will be included in a report documenting the issue. The report will be presented to the BFN PORC for approval prior to increasing reactor power.

The EPU testing program at BFN, which is based on the specific testing required for the BFN initial EPU power ascension, supplemented by normal Technical Specification testing and balance of plant monitoring, is confirmed to be consistent with the generic description provided in the CLTR.

5.0 Justification for Elimination of Power Ascension Tests

The justifications for not re-performing steady-state or small transient tests during EPU startup testing, that were performed at ≥ 80 percent OLTP during initial startup are summarized in Table 46-1.

For large transient tests that were performed during initial startup, and that will not be performed during EPU startup, the justification is summarized in Section 5.1. This applies to the following tests: No. 25, No. 27, No. 30, and No. 31.

5.1. Guidelines of SRP Section 14.2.1 Paragraph III.C.2

Paragraph III.C.2 of SRP Section 14.2.1 provides specific guidance to consider when justifying elimination of large scale transient testing. The following table provides a cross reference between the guidance of SRP Section 14.2.1 paragraph III C.2 and this attachment.

Table 5-1 Justification Cross Reference

SRP Section 14.2.1 Paragraph III C.2	Guidance Criteria	Discussion
(a)	Previous operating experience	Contained in paragraphs 5.3 and 5.4. Considers industry and BFN operating experience.
(b)	New thermal hydraulic phenomena or system interactions	No new thermal hydraulic phenomena or new system interactions were identified because of BFN EPU. No further discussion is provided.
(c)	Conformance with limitation of analytical methods.	BFN has no unique limitations associated with analytical methods. No analytical results are used as the sole justification for eliminating any tests. No further discussion is provided.
(d)	Plant staff familiarization with facility operation and EOPs.	Discussed in Section 5.2, 5.4.1 (MSIV Closure Test), and 5.4.2 (Generator Load Rejection Test).
(e)	Margin reduction in safety analysis for Anticipated Operational Occurrences (AOOs)	Provided in Section 5.4 for specific tests as applicable and in the section on EPU analysis results.
(f)	Guidance in Vendor topical reports	Discussed in Section 2 and 5.1.
(g)	Risk implications	Discussed in Section 5.2.

ELTR1 (Section 5.11.9) states MSIV closure test should be performed for EPU if the power uprate is more than 10% above any previously recorded MSIV closure transient. ELTR1 also states a generator load rejection test should be performed if the uprate is more than 15% above

any previously recorded generator load rejection transient. ELTR1 applies to extended power uprates whether constant pressure or otherwise.

With regard to the specific ELTR1 requirements for Large Transient Testing, BFN had these actual plant transients in the past several years. Unit 1 performed a MSIV Closure as part of the Unit 1 Restart Test Program. Unit 2 had a MSIV closure event at 100% power on June 9, 2010. Unit 3 has not had a MSIV closure event since initial power ascension. Unit 1 had a generator load reject at 100% power on February 18, 2009. Unit 2 had a generator load reject at 100% power on July 27, 2002. Unit 3 had a generator load reject at 100% power on September 28, 2011. Based on these events, the ELTR1 criteria to perform testing would apply to BFN as shown below in Table 5-2.

Table 5-2 Transient Testing Applicability

Event	Unit	LER No.	Event Date	Event Power	EPU Power	Percent Increase	Required by ELTR1
MSIV Closure	1	N/A (Restart Test)	06/23/2007	3458 MWth	3952 MWth	14.3%	Yes (EPU > 10%)
MSIV Closure	2	50-260/ 2010-003-00	06/09/2010	3458 MWth	3952 MWth	14.3%	Yes (EPU > 10%)
MSIV Closure	3	N/A (Initial Startup Test)	12/03/1976	3178 MWth	3952 MWth	24.0%	Yes (EPU > 10%)
Generator Load Rejection	1	50-259/ 2009-001-00	02/18/2009	3458 MWth	3952 MWth	14.3%	No (EPU < 15%)
Generator Load Rejection	2	50-260/ 2002-002-00	07/27/2002	3458 MWth	3952 MWth	14.3%	No (EPU < 15%)
Generator Load Rejection	3	50-296/ 2011-003-01	09/28/2011	3458 MWth	3952 MWth	14.3%	No (EPU < 15%)

TVA takes exception to the ELTR1 criteria for MSIV closure testing, and provides the below justification for elimination of large transient testing from the BFN EPU power ascension test plan.

5.2. Justification for Not Performing Large Transient Testing

Prior Large Transient Testing

Large Transient Testing performed during initial plant startup determined integrated plant response after reaching full power. Startup tests were required to baseline plant responses and to validate individual system performance. Startup test results indicate SSCs perform their intended functions. BFN Units 1, 2, and 3 either initially satisfied acceptance criteria, or resolved deficiencies and retested issues to satisfy all acceptance criteria. Large transient tests were not performed on Units 2 and 3 for the five percent stretch uprates in 1998. As part of the 2007 Unit 1 Restart Test Program, some transient tests were performed at 105 percent OLTP. Further Large Transient Testing for EPU is not required because plant transient performance has been baselined by startup testing, actual events, post modification testing and by analytical techniques.

Minimal Gains from Large Transient Testing

Large Transient Testing provides information that has minor additional value for plant operation. Large Transient Testing challenges a limited number of systems and components, all of which have a history of safe performance at BFN Units 1, 2, and 3. BFN Units 1, 2, and 3 have accumulated more than 80 operating years of experience dealing with plant transient response. Therefore, performance of additional testing to demonstrate plant response at EPU provides insignificant benefit.

No new transients occur as a result of EPU. Transient analyses at EPU are comparable to analyses at current plant conditions. Changes in plant conditions for EPU are not expected to result in a significant change to current plant conditions and transient response. Therefore, large transient testing at BFN will not provide new insights and any gains from this testing are minimal.

The benefits from Large Transient Testing are outweighed by the potential adverse affects Large Transient Testing has on plant equipment. Large Transient Testing has a negative impact on the station and the power grid, for which each unit supplies a significant base load. The scram and subsequent rapid reduction in power is controlled by normal operator procedural actions. Therefore, the requirement to perform Large Transient Testing at BFN to demonstrate safe operation of the plant is unwarranted.

No new thermal-hydraulic phenomena or system interactions have occurred following actual MSIV closure, turbine trip and load reject events at BFN. The plant has responded as expected in agreement with design features.

The proposed EPU test program will be included in a master test instruction which tests, sequences, and coordinates the aggregate impact of EPU, plant modifications, and normal refueling test program. Plant modifications to support EPU have minimal safety significance and will be implemented and tested as needed in advance of ascending from CLTP to EPU power.

BFN Simulator Models Transients

Advances in analytical techniques, methods, models, and simulators have created a high level of confidence in determining plant responses and are cost effective alternatives to actual testing. Analyses demonstrate that plant shutdown is safely achieved under EPU conditions. TVA will perform simulator demonstrations of plant transient performance to support operator training at EPU conditions.

As mentioned previously, BFN Units 1, 2, and 3 have a similar system geometry, reactor protection system, configuration and mitigation features. Additionally, BFN Units 1, 2, and 3 have similar thermal-hydraulic and transient behavior characteristics. BFN could train with one common simulator, but BFN has two simulators to support the logistics of training the large staff associated with three units. The plant simulators are based on Unit 2 and Unit 3. Both simulators are benchmarked against their respective units. The simulators provide accurate NSSS and BOP modeling of transients. This modeling of the units facilitate operator training on plant response to potential transients or events. At least one simulator will be updated to model EPU operation including all modifications and transient analyses prior to EPU implementation on the lead unit. This simulator will be used for operator training prior to EPU implementation. BFN operators train on various plant upset conditions from postulated accident conditions to anticipated transients. This training prepares them for the type, timeline, and extent of the plant response to transients. Therefore, initiating actual plant transient events for purposes of operator training will not be necessary.

Large Transient Testing Risk Assessment

TVA conducted an EPU probabilistic risk assessment (Attachment 44). It includes an assessment of performing two plant transient tests upon BFN EPU implementation. The evaluated tests were a generator full load reject and an MSIV isolation event. The generator full load reject test was evaluated by simulating the turbine trip event, and the MSIV isolation test was evaluated by simulating an MSIV closure event. The risk assessment indicated the proposed tests represent an increase in the risk of core damage and large early release. This assessment does not include the potential equipment damage or challenges to the operators, which should be avoided. The CCDPs and CLERPs for a turbine trip and for a MSIV closure event are relatively small; however, they do have some risk significance.

For perspective on the risk incurred by performing these tests, it is beneficial to compare the risk against the risk acceptance guidelines in Regulatory Guide 1.174, which discusses probabilistic risk assessment and risk-informed decisions. Although the figures of merit in RG 1.174 are ΔCDF and ΔLERF, the CCDP and LLERP can be compared directly with ΔCDF and ΔLERF assuming a frequency of once per year. Comparing the CCDP and CLERP for the MSIV event in Table 5-3 with Figure 4 and Figure 5 in RG 1.174, the values are very near the boundary between Region I and Region II. If a nuclear power plant proposed performing an annual surveillance test that placed the plant in Region I, RG 1.174 would suggest that this change would be unacceptable.

Table 5-3 Conditional Probabilities for BFN EPU Startup Testing

Unit	Initiating Event	Conditional Core Damage Probability (CCDP)	Conditional Large Early Release Probability (CLERP)
Unit 1	Turbine Trip	2.10E-07	3.88E-08
Unit 1	MSIV Closure	9.42E-06	2.20E-06
Unit 2	Turbine Trip	1.92E-07	3.59E-08
Unit 2	MSIV Closure	9.40E-06	2.20E-06
Unit 3	Turbine Trip	1.61E-07	3.15E-08
Unit 3	MSIV Closure	8.11E-06	2.00E-06

Note: These results are based on Internal Events PRA.

These CCDPs and CLERPs represent the incurred risk caused by performing the proposed tests (i.e., the initiating events occur). If both tests are performed, the total conditional probabilities would be for Unit 1: 9.63E-6 (CCDP) and 2.24E-06 (CLERP), for Unit 2: 9.59E-6 (CCDP) and 2.24E-6 (CLERP), and for Unit 3: 8.27E-6 (CCDP) and 2.03E-06 (CLERP). Note the analyses do not credit compensatory measures that may reduce the risk of core damage given that extra operators may be staged for the proposed tests. .

5.3. Post EPU Industry Operating Experience

Steam Dryer Issues

Stresses imposed on steam dryers by the higher steam flows are being addressed in Attachment 40 of the BFN EPU application; therefore, will not be repeated here.

Industry Post EPU Transient Events

Several BWR-4 plants (similar to BFN) have completed an EPU uprate. A review of industry transient events that occurred after NRC approval of the CLTR in March 2003, at greater than original power levels, was performed. Several examples of BWR-3/4/6 plant responses to MSIV closure and load reject/turbine trip events are detailed in the examples below. As indicated, the plants responded as expected in accordance with their design features. No unexpected conditions were experienced nor were any latent defects uncovered in these events beyond the specific failures that initiated the events. These events provide further evidence that large transient testing is unnecessary.

<u>Susquehanna Steam Electric Station (BWR 4) - 20% Approved Power Uprate</u>

LER 50-388/ 2011-003 (Main Turbine Trip)

On August 19, 2011, at 10:46 hours, Susquehanna Steam Electric Station (SSES) Unit 2 automatically scrammed from 100 percent power due to a main turbine trip. The main turbine trip occurred during the performance of the quarterly functional surveillance test of the reactor water-high-level trip channels for feedwater / main turbine. The surveillance test was being performed for the first time since the 2011 upgrade of the Unit 2 feedwater level control system with a digital Integrated Control System (ICS). As part of the test, operations personnel transferred the reactor water level input signal from average level to narrow range 'B' biased as required by the procedure. The main turbine and feedwater-trip system design uses three narrow range reactor water level channels in a two-out-of-three trip logic. When the first narrow range reactor water level channel (2A) was tested, an unexpected automatic main turbine trip occurred. An investigation revealed an internal jumper was incorrectly terminated. The wiring anomaly in the ICS level 8 turbine-trip-logic circuitry resulted in one of the level 8 trip logic contacts being "jumpered-out" of the channel trip circuitry, causing a Unit 2 main Turbine Trip from the initiation of one single channel instead of the designed two-out-of-three channel logic.

Actual reactor vessel water level was within the normal band when the main turbine tripped. The main turbine trip resulted in a reactor scram and the reactor recirculation pump trips as designed. All control rods fully inserted. Reactor water level lowered to +2 inches causing Level 3 (+13 inches) isolations.

Reactor water level was restored to normal operating band using the feedwater system. Level setpoint setdown to +18 inches occurred as designed. Due to loss of forced core recirculation flow and potential for reactor thermal stratification, control rod drive system flow was reduced and reactor water level was raised to provide natural circulation flow. The reactor recirculation pumps were subsequently restarted to re-establish forced core circulation. The maximum differential temperature observed between the bottom head region and bulk coolant temperature was 92 degrees Fahrenheit, which was within the 145 degree Fahrenheit pump start limit.

Six main steam relief valves (SRVs) opened for a short duration as expected due to the turbine trip transient. The main steam isolation valves (MSIV's) remained open during the transient. Reactor pressure was controlled via turbine bypass valve operation. A management decision to not initiate a forced cooldown was made, and reactor pressure slowly lowered due to insufficient core decay heat to maintain normal operating pressure. The gradual cooldown did not challenge the 100 degree Fahrenheit [per hour] cooldown limit. No Emergency Core Cooling Systems (ECCS) or Reactor Core Isolation Cooling (RCIC) system initiations occurred or were required.

Hope Creek Generating Station (BWR 4) - 15% Approved Power Uprate

LER 50-354/ 2013 -008 (Main Turbine Trip)

On December 1, 2013, at 06:13 EST, Hope Creek Unit 1 automatically scrammed from 100 percent rated thermal power due to a trip of the main turbine. The main turbine trip was due to a high level in the 'A' moisture separator. The main turbine trip caused an actuation of the reactor protection system resulting in an automatic reactor scram. Both reactor recirculation pumps tripped per design and three safety relief valves (SRV) lifted. The plant was stabilized in hot shutdown. All control rods inserted as required and no automatic emergency core cooling system (ECCS) or reactor core isolation cooling (RCIC) system initiations occurred.

Edwin I. Hatch Nuclear Plant (BWR 4) - 15% Approved Power Uprate

LER 50-321/ 2005 -002 (Main Generator Trip)

On 10/29/2005 at 1330 EST, Unit 1 was in the Run mode at a power level of 2804 MWT (115 OLTP). At that time, the reactor automatically tripped on turbine control and stop valve fast closures caused by main turbine and main generator trips. The turbine and generator tripped when the main power transformer experienced a fault, which resulted in a main generator neutral ground overcurrent lock-out. The main power transformer is a generator step up transformer, 24kv to 230kv. Actuation of this lockout generated direct turbine and generator trip signals and the main turbine and generator tripped per design. These trips resulted in fast closure of the turbine control and stop valves. Fast closure of either the turbine control valves or stop valves is a direct input to the reactor protection system.

Following the automatic reactor trip, vessel water level decreased due to void collapse from the rapid increase in reactor pressure. Water level reached a minimum of approximately 16 inches below instrument zero (about 142 inches above the top of the active fuel) resulting in closure of the Group 2 primary containment isolation valves. The operating reactor feedwater pump recovered reactor vessel water level, restoring level to between 23 and 48 inches above instrument zero for the remainder of the event. Level did not decrease to the point of Group 1 isolation. Therefore, the main steam isolation valves remained open throughout the event.

Vessel pressure reached a maximum value of 1145 psig after receipt of the reactor trip. This pressure is within the band of the electronic actuation setpoints as well as the pressure relief setpoints for the safety/relief valves (S/RVs). Consequently, all eleven of the S/RVs actuated properly to reduce reactor pressure. The Low-Low Set function armed and initially operated to reduce reactor pressure and controlled reactor pressure down to 847 psig. Since the main steam isolation valves remained open, the main turbine bypass valves functioned to control vessel pressure thereafter.

LER 50-366/ 2006-002 (Main Turbine Trip)

On April 5, 2006, Hatch Unit 2 was operating at 100% (115% OLTP) rated thermal power when a power-load unbalance was sensed resulting in a Turbine Control Valve fast closure and subsequent reactor scram. Reactor pressure spiked to approximately 1,125 psig, which resulted in eight of the eleven Safety Relief Valves opening to relieve reactor pressure. Vessel water level was maintained well above the top of the active fuel throughout the transient and never decreased to the reactor scram actuation setpoint. Reactor water level was maintained through the use of the reactor feed pumps and manual initiation of the RCIC and High Pressure Coolant Injection (HPCI) systems. There were no automatic safety system actuations on low level.

LER 50-321/ 2008-003 (Main Turbine Trip)

On July 4, 2008, Hatch Unit 1 was at 99.7% (115% OLTP) rated thermal power and experienced a turbine trip during testing of the Electrohydraulic Control (EHC) system. The resultant Turbine Control Valve fast closure initiated a reactor scram, as designed. Following the reactor scram, reactor pressure peaked at approximately 1,120 psig, resulting in four of the eleven Safety Relief Valves opening as designed to reduce pressure. The feedwater level control system controlled reactor water level with a minimum water level of approximately 2.5 inches above instrument zero (about 160 inches above the top of active fuel). All required safety systems functioned as expected given the water level and pressure transients caused by the turbine and reactor trips. Vessel water level was maintained well above the top of active fuel throughout the transient.

<u>Clinton Power Station (BWR 6) - 20% Approved Power Uprate</u>

LER 50-461/ 2013 -002 (Main Generator Trip)

At 0642 on 3/7/13, the plant was in Mode 1 (Power Operation) at 96.9 percent power. The Main Control Room (MCR) received a main generator trouble alarm for the Automatic Voltage Regulator (AVR) automatically transferring from channel 2 to channel 1 due to a fault. Operators verified main generator parameters were normal and dispatched Equipment Operators and Electrical Maintenance technicians to investigate the reason for the fault. The MCR completed pre-emptive briefs to discuss the alarms received and a Reactor Operator was assigned to focus on the main generator and exciter Contingency actions for a turbine trip and reactor Scram were also briefed.

At 0756 hours, the main generator tripped. At 0758 hours, Operators in the MCR received numerous alarms for a main turbine trip and reactor Scram. Subsequently, the Reactor Operator placed the reactor mode switch into the shutdown position. Operators entered the Reactor Scram Off-Normal Procedure and subsequently entered Emergency Operating Procedure (EOP)-1, "Reactor Pressure Vessel Level Control," due to an expected low reactor water level 3 trip signal. All control rods fully inserted and all plant equipment responded as expected to the Scram. At 0928 hours, Operators established a reactor coolant pressure of 500 to 600 psig, using Turbine Bypass Valves, a reactor pressure vessel water level of 30 to 39 inches, and exited EOP-1 in accordance with normal plant procedures.

As expected during the event, the low reactor water level 3 trip signal caused primary containment isolation valves in Group 2 (Residual Heat Removal (RHR)), Group 3 (RHR), and Group 20 (miscellaneous systems) to receive signals to shut; operators verified that the valves properly responded to the trip signal.

<u>Grand Gulf Nuclear Station (BWR 6) - 15% Approved Power Uprate</u>

LER 50-416/ 2013 -002 (Main Generator Trip)

At 18:05 Central Standard Time on January 14, 2013, Grand Gulf Nuclear Station experienced an automatic reactor Scram caused by a Turbine Trip due to Main Generator lockout. The plant was operating in Mode 1 at 100 percent thermal power. All safety systems responded per design. Safety Relief Valves (SRVs) opened at the onset of the event to control reactor pressure and reseated properly. All control rods inserted when the signals generated by the RPS were received. There were no Emergency Core Cooling System actuations. The shift immediately entered the appropriate Off Normal Event Procedures and Emergency Procedures. The plant was stabilized with pressure control on the main turbine bypass valves and level control on the start-up level control valve. High pressure feedwater heater start-up outlet valve (start-up outlet valve) did not open when placing the start-up level control valve in service but did not prevent Operations from controlling the reactor water level. The plant responded to the trip as designed with the exception of the one start-up outlet valve noted above.

<u>Dresden Nuclear Power Station (BWR 3) - 17% Approved Power Uprate</u>

LER 50-237/ 2005-002 (MSIV Closure)

On March 24, 2005, at 0529 hours (CST), with Unit 2 at approximately 96 percent power, two unexpected control room alarms were received for exceeding the Electro-Hydraulic Control System maximum combined flow limit setpoint and open Turbine Bypass Valves. Several seconds later, high flow in the Main Steam System resulted in a signal to close the Main Steam Isolation Valves that initiated an automatic reactor scram. All control rods fully inserted and all other systems responded to the reactor scram as expected, except for non-safety related equipment, the Turbine Generator Lube Oil Pump and the 2B Reactor Feedwater Pump Auxiliary Oil Pump, which did not operate as required.

LER 50-237/ 2006-004 (MSIV Closure)

On July 4, 2006, at approximately 0259 hours (CDT), with Unit 2 at approximately 98 percent power, the plant received a Group I Isolation on a Main Steam Line High Flow signal. The Main Steam Line High Flow signal was caused by the unexpected closure of the 1A Main Steam Isolation Valve (MSIV) and the resulting redistribution of steam to the remaining three main steam lines. All Group I isolation valves closed and the reactor automatically scrammed. All control rods inserted. Reactor water level was automatically controlled and the reactor vessel level shrink resulted in an expected Group II and Group III isolation signals. All systems responded as required and the Isolation Condenser was manually initiated to control reactor pressure.

The industry operating experience cited above indicates that other BWRs that have performed EPUs have successfully responded as planned to MSIV closure and load reject/turbine trip events. Based on this experience of other similar plants it is likely that BFN will perform similarly to these transients.

5.4. BFN Large Transient Testing Review and Analysis

5.4.1. MSIV Closure Test

Initial Startup Test Objectives and Results

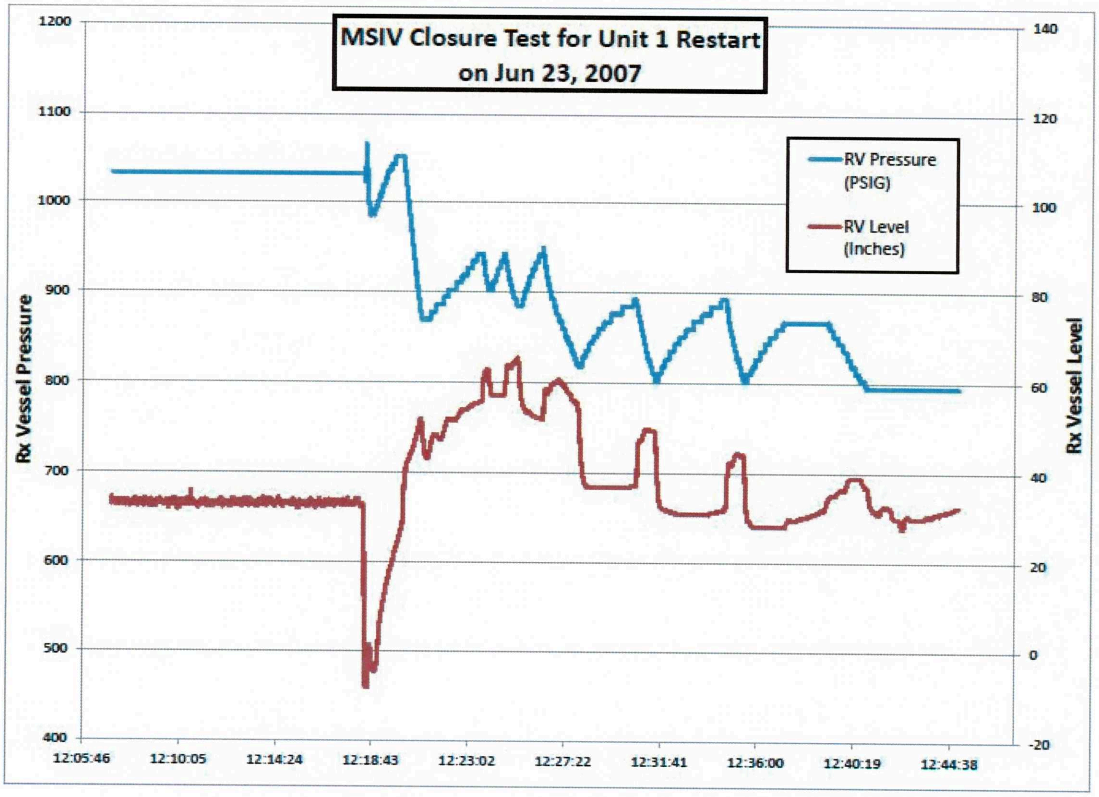
During the initial startup test program, this test (a) functionally checked the main steam line isolation valves (MSIVs) for proper operation at selected power levels, (b) determined reactor transient behavior during and following simultaneous full closure of all MSIV's and following full closure of one valve, (c) determined isolation valve closure time, and (d) determined the maximum power at which a single valve closure can be made without scram. A discussion of the test and acceptance criteria are contained in the BFN UFSAR Section 13.5.2 and 13.5.3 (Test No. 25).

All acceptance criteria for MSIV closure startup testing were satisfied. Proper MSIV operation was demonstrated and closure times verified at various power levels. Deficiencies were resolved and retested to verify acceptance criteria. During startup testing, MSIVs were closed and tested individually during heatup, 50% and 70% power plateaus. Proper operation was demonstrated and closure times were within limits. Reactor response was monitored and steam flow margins calculated and all results were within limits. For Units 2 and 3, a full closure of all MSIVs was initiated from approximately 100% power. For Unit 1, a full closure of all MSIVs was only initiated during Heatup. MSIV closure times and reactor parameters were monitored and found acceptable when compared to predicted results.

Unit 1 Restart Test Program and Results

For the Unit 1 restart, the NRC required a MSIV Closure Test. This test validated that upon full closure of all MSIVs the response of NSSS maintains vessel pressure below safety limit, and demonstrated proper operation of RCIC, HPCI, and the safety relief valves.

This large transient test was performed from 3456 MWth (104.9 percent OLTP). All acceptance criteria were met. Reactor steam dome pressure was maintained below 1230 psig following closure of all MSIVs. MSIV closure times were between 3 and 5 seconds; RCIC and HPCI auto started per design to restore vessel level. The short-term reactor pressure and level responses are shown below.



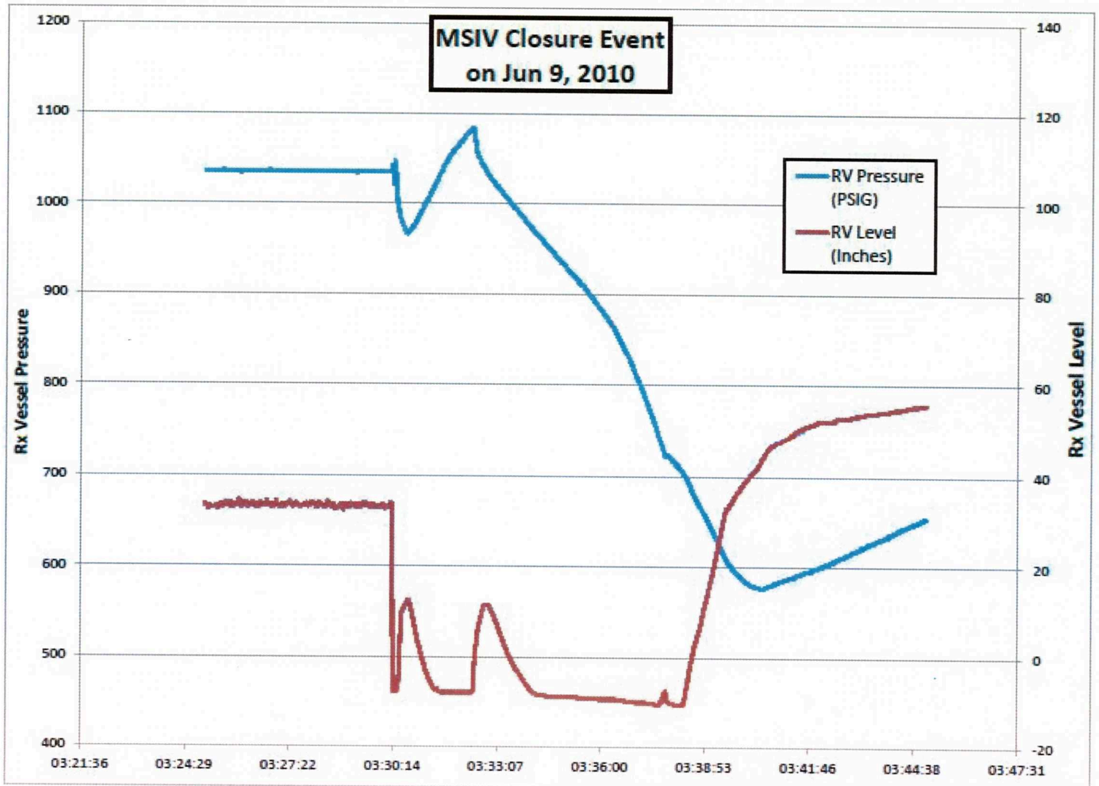
Previous BFN Operating Experience

LER 50-260/ 2010-003 (BFN Unit 2 MSIV Closure)

On June 9, 2010, at approximately 0330 hours Central Daylight Time (CDT) with Unit 2 operating at approximately 100 percent power (105 percent OLTP), the outboard Main Steam Isolation Valve (MSIV) A closed while transferring the Reactor Protection System (RPS) 120 V-AC power from the normal to the alternate power supply in preparation for a planned activity. At approximately 0331 hours CDT Unit 2 received a Primary Containment Isolation Signal (PCIS) Group 1 isolation signal resulting in the closure of all of the MSIVs and automatic reactor scram. During the scram, all automatic functions occurred as expected. All control rods inserted. Operations personnel briefly entered Emergency Operating Instruction, 2-EOI-001, "Reactor Pressure Vessel Control," controlling both reactor vessel pressure and reactor vessel water level. At approximately 0335 hours CDT, Operations personnel reset the PCIS Group 1 Isolation Signal, and by approximately 0341 hours the reactor scram was reset.

The Group 1 PCIS initiation signal was the only isolation signal received prior to the reactor scram. The A Control Room Emergency Ventilation (CREV) system auto initiated. Standby Gas Treatment (SGT) subsystems A, B, and C were in service prior to the event and continued to operate through the event. Operations personnel manually initiated High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems to control reactor water level. Reactor pressure vessel pressure was controlled by manually opening one safety relief valve and the MSL drain valves. At approximately 0335 hours CDT, Operations personnel reset the PCIS Group 1 Isolation Signal, and by approximately 0341 hours the reactor scram was reset. By 0405 hours CDT the MSIVs were reopened, Operations personnel then controlled the reactor pressure with the turbine bypass valves. A heat rejection path was established using the main condenser. HPCI and RCIC were removed from service and the reactor water level was being maintained with the condensate and feedwater systems.

The short-term reactor pressure and level responses are shown below.



The plant response following the MSIV closure transient was in accordance with expectations. This event was bounded by the transient analysis (Generator Load Reject Without Bypass Valves) as described in UFSAR Section 14.5.2 and UFSAR Appendix N.

<u>Plant staff familiarization with facility operation and EOPs</u>

The EPU will not change any plant operations or EOP actions associated with MSIV closure. Since the dome pressure does not change, SRV set point changes are not required. As discussed in PUSAR Section 2.2.2.2.1.2, the MSIVs have design features that ensure that MSIV closure time is maintained. The MSIV closing times will be tested and readjusted as necessary to ensure that the MSIV closure times satisfy BFN Technical Specifications. These minor changes do not alter the operation of the plant to address MSIV closure with increased steam flow.

Margin reduction in safety analysis for Anticipated Operational Occurrences (AOOs)

The EPU ASME Over-Pressurization Analysis is discussed in FUSAR Section 2.8.4.2. The analysis indicates that the predicted peak dome pressure for limiting MSIV events increases for EPU. The event was conservatively analyzed. Some of the conservatism were initial power level of 102% EPU RTP, initial dome pressure of 1055 psig, one relief valve assumed out of service, assumed relief valve setpoint drift of 3% plus + 5 psi uncertainty allowance, and no credit was taken for Scram from MSIV position. Calculated dome pressure increased approximately 20 psi to 1320 psig and vessel bottom pressure increased to 1349 psig. The margin to the dome pressure safety limit of 1325 psig is 5 psi, and the margin to the vessel bottom pressure safety limit of 1375 psig is 26 psi.

Results demonstrate the maximum vessel pressure limit of 1375 psig and dome pressure limit of 1325 psig are not exceeded. The peak pressure results include adjustments to address the NRC concerns associated with the void-quality correlation, exposure-dependent thermal conductivity, and Doppler effects. This analysis is updated each fuel reload. The EPU analyses conclude that adequate margins exist to accommodate cycle specific variances and to ensure that all ASME Code requirements continue to be satisfied.

EPU Power Ascension Testing

MSIV full closure testing at 100% rated power during EPU power ascension testing is not required at BFN because the plant response at EPU conditions is expected to be similar to the documented response during initial startup testing, Unit 1 Restart, and actual transients that have occurred during plant operation. The transient analysis performed for the BFN EPU demonstrates that all safety criteria are met, and for EPU that the MSIV closure event is limiting.

Deliberately closing all MSIVs from 120% OLTP power will result in an undesirable transient cycle on the primary system that can reduce equipment service life. As demonstrated during initial startup or restart testing and confirmed by analysis, all equipment responses to the transient are within component and system design capabilities. However, placing accident mitigation equipment into service, under maximum loading conditions, uses available service life. Equipment service life should be retained for actual events rather than for demonstration purposes. Additional transient testing and the resulting impact will provide no additional plant response information beyond that documented during previous testing and from the evaluation of actual plant events. These events demonstrate the analysis is conservative and actual events will not challenge safety or design limits for this event.

The MSIV modifications (internal changes to reduce the pressure drop across each valve at EPU steam flow) will not have an impact on this event because correct valves functions are confirmed by the In-Service Testing (IST) Program. This program establishes the testing and examination requirements to assess operational readiness of the MSIVs. Per the IST Program, single full MSIV closures are not performed at power at BFN. Only partial MSIV stroke tests are performed to verify associated RPS functions. Fast closure full-stroke time testing of the MSIVs occurs during refueling outages.

The modifications to the main steam, feedwater, and condensate systems will not negatively influence this event. Surveillance testing will ensure proper operation of the MSIVs. Planned EPU Power Ascension Testing (Test No. EPU-23) will ensure proper feedwater water level control. Operational level control strategies and pressure control strategies maintain margin to the level 8 trip setpoint, which causes the main turbine, feedwater turbine, HPCI pump turbine, and RCIC pump turbine to trip under CLTP and EPU conditions. The level 8 isolation signal provides the margin necessary to ensure reactor water level will not approach the elevation of the main steam lines.

The modification to the main turbine and the reduction in bypass capability has no impact because the turbine and the bypass system are isolated for this event.

Conclusion

TVA has reviewed the initial startup and restart testing, recent BFN and industry operating experience, as well as, analysis and PRA results. On Units 2 and 3, the original startup MSIV closure test was performed at 100% OLTP (3293 MWt). On Unit 1, the plant response to a MSIV closure event was demonstrated at 100% CLTP (3458 MWt) which is equivalent to 88% EPU (3952 MWth). Based on plant historical data and EPU analytical results, it is concluded that the MSIV Closure Event results in conditions that are within design limits.

No new design functions in safety related systems are introduced as part of EPU that would need large transient testing validation. No physical modifications or setpoint changes are made to the pilot operated SRVs.

The increase in main steam flow and its impact is not significant with regard to the reactor pressure transient response. The modifications to the feedwater and condensate systems do not adversely change the feedwater level control response and the use of RCIC as the preferred level control system for this event.

In view of the above, the objective of determining reactor transient behavior resulting from the simultaneous full closure of all MSIVs can be satisfied through analysis for EPU and without large transient testing. In addition, limiting transient analyses are included as part of the cycle specific reload licensing analysis. The need for re-performing this test at EPU conditions is not required because plant response is not expected to significantly change from that previously documented at CLTP conditions. Plant performance and analysis show adequate margin is available in vessel pressure and level limits to demonstrate acceptable reactor transient behavior. Therefore, this test is not warranted.

5.4.2. Generator Load Rejection Test

Initial Startup Test Objectives and Results

The startup testing for generator load rejection demonstrates the transient performance of the reactor and the station electrical supply system following a generator trip. A description of these tests and acceptance criteria are contained in the BFN UFSAR Section 13.5.2 and 13.5.3 (Test No. 27 and Test No. 31).

The initial Generator Load Rejection Tests for BFN were performed at 25% power and included loss of all offsite power (Test No. 31). Additional startup testing of turbine stop and control valve closure/ generator trip were performed at power levels up to 100% OLTP (Test No 27). Tests verified reactor transient responses and compliance with acceptance criteria. Deficiencies were diagnosed, resolved, and retested, as required, to verify acceptance criteria. These startup tests, and recent BFN and industry operating experience, at increased power levels, demonstrate adequate plant response to this event and further testing is not considered necessary.

<u>BFN Operating Experience</u>

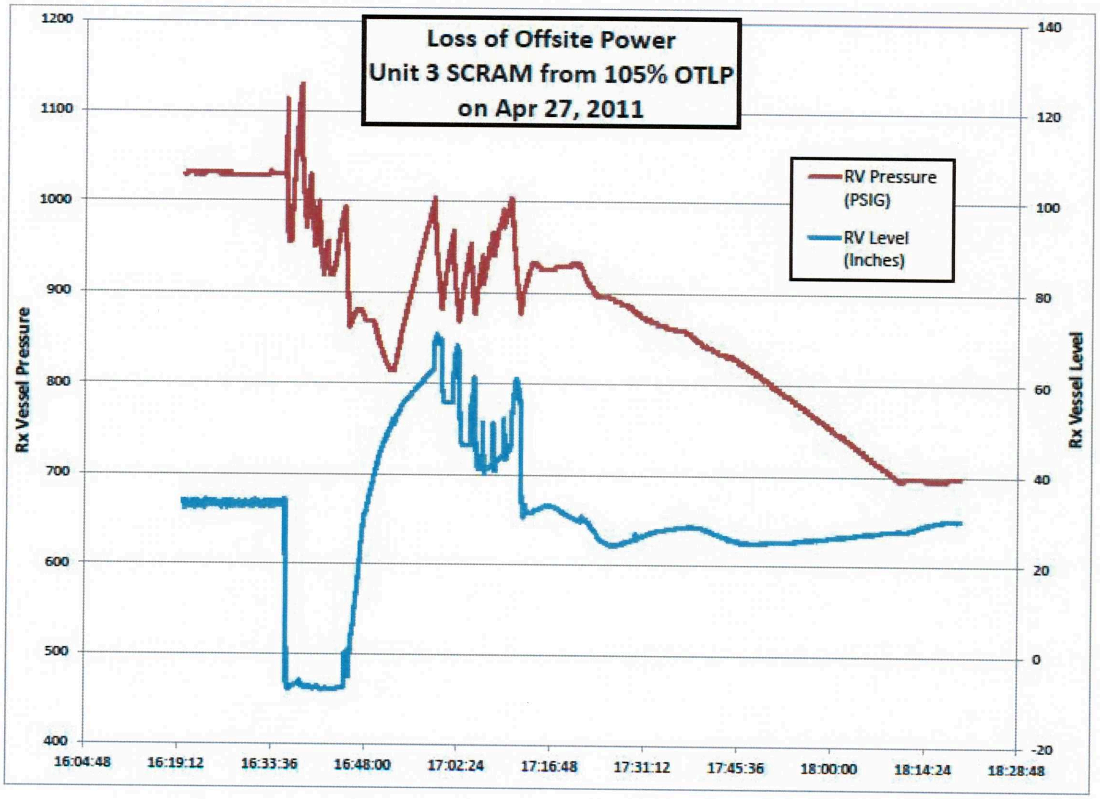
LER 50-259/ 2011-001 (BFN Units 1, 2, & 3 Scram Caused by Loss of All 500-kV Offsite Power Sources)

On April 27, 2011, following offsite power grid oscillations (due to severe weather including high winds and tornadoes) BFN experienced a complete loss of the 500-kV offsite power system. This resulted in automatic scrams of Units 1, 2, and 3. All three units were in Mode 1 prior to the event. Units 1 and 2 were at 75 percent power, and Unit 3 was 100 percent power (105 percent OTLP). All scram systems were actuated, all actuations were complete, and required systems started and functioned successfully with the exception of an indeterminate position indication for the Unit 3 B Inboard Main Steam Isolation Valve (MSIV). All onsite safe shutdown equipment was available with the exception of the 3B Emergency Diesel Generator (EDG), which was inoperable and unavailable due to planned maintenance. After the event, only one 161-kV line remained available for offsite power - all (seven) 500-kV lines and one (of two) 161-kV line were de-energized. All three units immediately entered Mode 3 (Hot Shutdown) with their respective 4-kV busses supplied from the onsite EDGs.

Following the automatic scrams, Operations personnel used the applicable post-scram procedures. Unit-specific emergency operating procedures were utilized because the scrams were complicated by the loss of normal power to balance of plant systems.

On April 28, 2011, all three units entered Mode 4 (Cold Shutdown). On May 2, 2011, all shutdown boards were powered from qualified 161-kV offsite power sources, and all EDGs were shutdown and placed in standby readiness.

The short-term reactor pressure and level responses for Unit 3 (which started event from 105 percent OLTP) are shown below.



LER 50-259/ 2009-001 (BFN Unit 1 Main Generator Trip)

February 18, 2009, at 0351 hours Central Standard Time (CST), a Unit 1 Main Generator Trip, Main Turbine Trip, and Reactor Scram occurred from 100 percent power (105 percent OLTP). Unit 1 reactor automatically scrammed from a turbine trip due to a power load unbalance signal on the main generator. Specifically, at 0349 hours CST, Operations swapped the Unit 1 Main Generator Isophase Bus Duct System cooling fan from the running to the alternate fan. When the alternate fan started, water entrapped in the fan housing was expelled into the bus providing a path to ground inside the bus duct. This resulted in actuation of the generator protective relays, a turbine trip, and automatic reactor scram. These actions resulted in the automatic actuation of the reactor protection system.

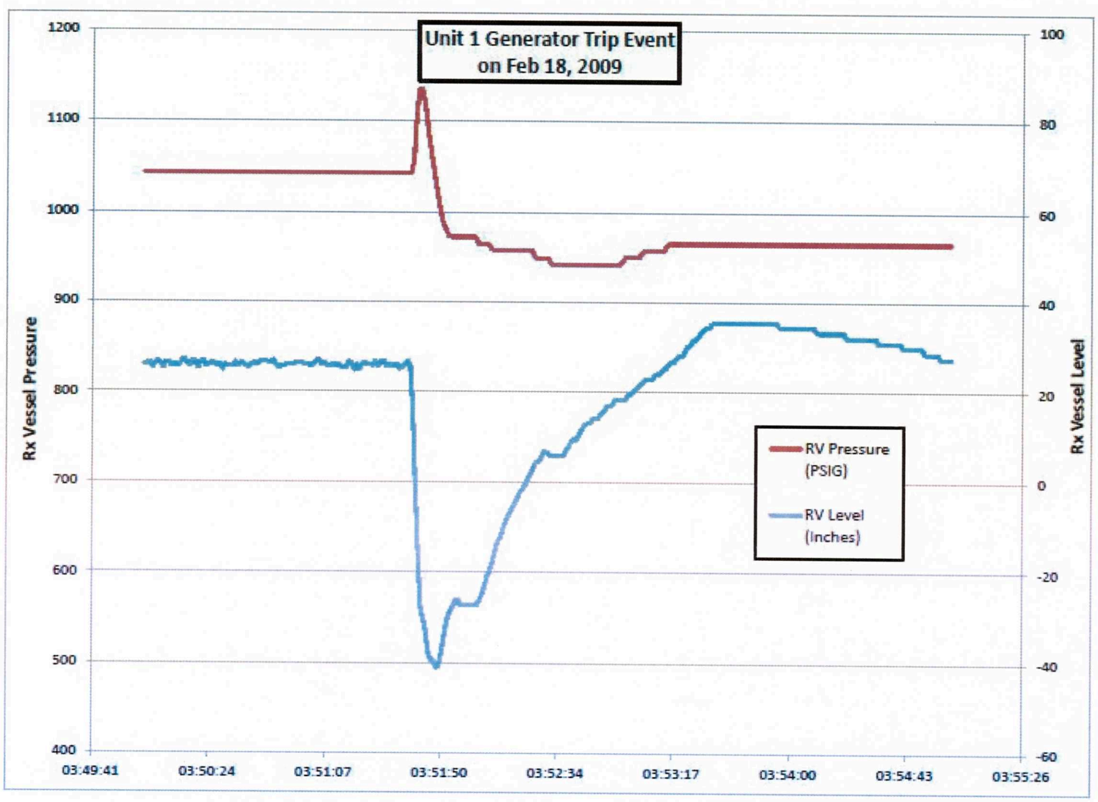
All automatic functions resulting from the turbine trip and automatic reactor scram occurred as expected. All control rods inserted. The level 3 scram setpoint was reached during the post scram water level shrink. Thus, the primary containment isolation system (PCIS) isolations: Group 2 (residual heat removal (RHR) system shutdown cooling), Group 3 (reactor water cleanup (RWCU)), Group 6 (ventilation), and Group 8 (traversing incore probe (TIP)) were received along with the auto start of the control room emergency ventilation (CREV) system and the three standby gas treatment (SGT) system trains. The reactor scram resulted in the reactor water level briefly attaining minus 43-inches, and reactor pressure 1140 psig, hence; Operations briefly entered Emergency Operating Instruction, (EOI-001) Reactor Pressure Vessel Control.

Following verification that the 1-AOI-100-1, Reactor Scram, actions were completed the reactor mode switch was placed in shutdown. Operations reset the reactor scram by 0420 hours CST. At approximately 0420 hours CST, operations reset the PCIS actuations and secured the SGT and CREV systems.

During and following the automatic scram, all safety systems operated as required. The operator actions taken in response to the scram were appropriate. These actions included the verification that the reactor had shutdown, the expected system isolations and indications had occurred, and subsequent restoration of these systems to normal pre-scram alignment.

PCIS groups 2, 3, 6, and 8 isolations were as expected. Although the Emergency Core Cooling Systems were available, none was required. No main steam relief valves actuated. The turbine bypass valves maintained reactor pressure. The main condenser remained available for heat rejection. Reactor water level was recovered and maintained by the reactor feed water and condensate systems.

The short-term reactor pressure and level responses are shown below.



LER 50-260/ 2002-002 (BFN Unit 2 Main Generator Trip)

On July 27, 2002, a Unit 2 Main Generator Trip, Main Turbine Trip, and Reactor Scram occurred from 100 percent power (105 percent OLTP). All expected system responses were received, including the automatic opening of four safety-relief valves. Actuation of primary containment isolation system groups 2, 3, 6, and 8 occurred due to the expected temporary lowering of reactor water level. This logic isolates shutdown cooling (if in service), isolates the reactor water cleanup system, isolates the normal reactor building ventilation, initiates the standby gas treatment and the control room emergency ventilation systems, and retracts traversing incore probes (if inserted). 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. Neither the high pressure coolant injection nor reactor core isolation cooling systems were used during this event.

Equipment response following the turbine trip and reactor scram was in accordance with the plant design. The short-term pressurization transient was mitigated by SRV and turbine bypass valve operation, and pressure control following the initial transient was handled by the bypass valves. The operation of other systems post-scram (e.g., containment isolation, start up of SGT and CREV, isolation of normal reactor building ventilation, RWCU isolation, TIP isolation, etc.) also occurred in accordance with the plant design. The main condenser continued to function as the heat sink following the scram. All operator actions were appropriate.

The generator tripped due to a ground fault on a main bank transformer bushing, which occurred due to thermal degradation of the paper insulation of the bushing’s internal condenser.

LER 50-296/ 2007-005 (BFN Unit 3 Main Generator Trip)

On December 31, 2007, at 2140 hours Central Standard Time (CST), Unit 3 reactor received an automatic scram signal following a main generator load reject from 100 percent power (105 percent OLTP). The reactor scram from the generator load reject was expected.

All systems responded to the scram as expected. All control rods inserted. Because the reactor level lowered to level 3 (low level) primary containment isolation system (PCIS) isolations Group 2 (residual heat removal (RHR) system), Group 3, (reactor water cleanup (RWCU) system), Group 6 (ventilation), and Group 8 (traversing incore probe (TIP) system) signals were received. The low water level also initiated the standby gas treatment (SGT) system and the control room emergency ventilation (CREV) system. The reactor water level remained above level 2 (low-low level); accordingly, no emergency core cooling systems were actuated. The reactor scram and PCIS actuations were reset December 31, 2007, by 2146 hours CST, SGT and CREV systems were secured by 2153 hours CST.

During the initial pressure transient, which peaked at 1141 psig, six (6) of the main steam system relief valves opened. The reactor pressure was subsequently controlled with the main steam system bypass valves. The reactor water level was controlled by the feedwater system; the normal heat removal path through the main condenser was maintained during the event.

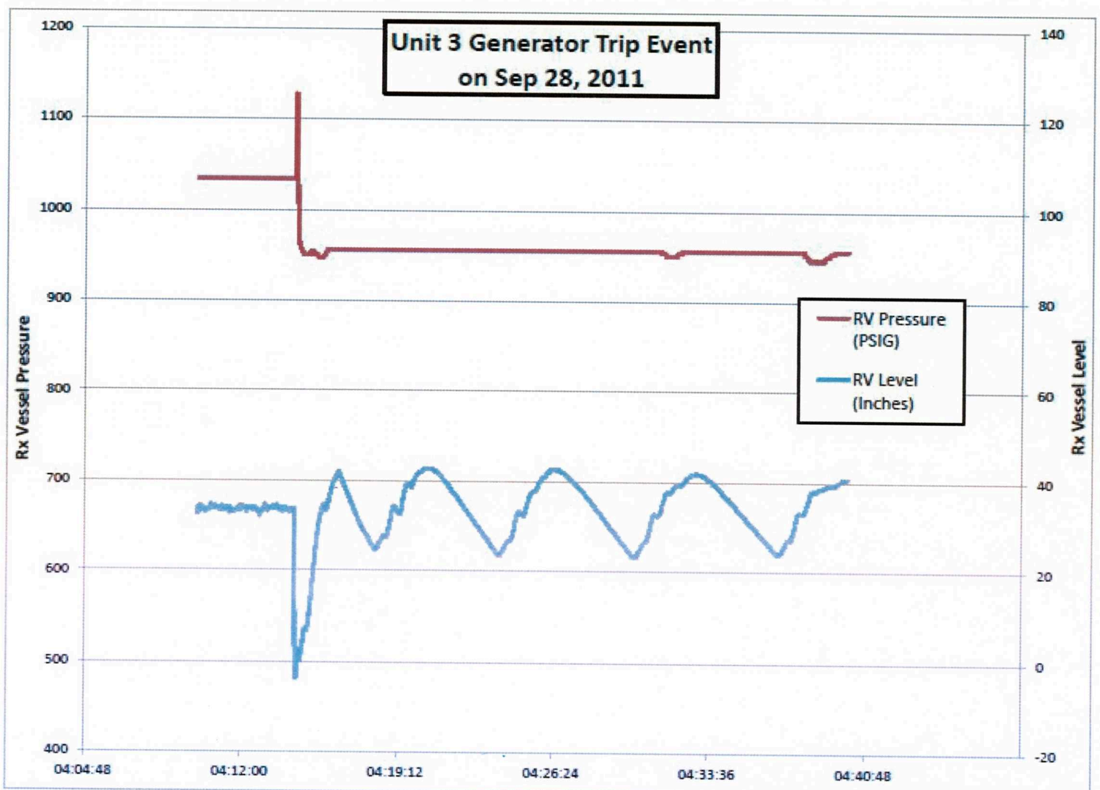
The generator trip was the result of spurious operation of the Unit 3 generator breaker phase discordance Relay (20-7 relay) which resulted in a generator breaker trip signal.

LER 50-296/ 2011-003 (BFN Unit 3 Main Generator Trip)

On September 28, 2011, at 0414 hours Central Standard Time (CST), Unit 3 reactor received an automatic scram signal following a main generator load reject from 100 percent power (105 percent OLTP). Seven safety relief valves (S/RVs) cycled due to the reactor pressure transient. All systems responded as expected to the turbine trip. There were no Low Pressure Coolant Injection System (LPCI), Core Spray System (CS), High Pressure Coolant Injection System (HPCI), or Reactor Core Isolation Cooling System (RCIC) reactor water level initiation set points reached. Primary containment isolation and initiation signals from groups 2, 3, 6, and 8 were received as expected. Reactor water level was automatically controlled by the Feedwater System.

The immediate cause of this event was a piece of the Iso-Phase Bus (IPB) debris screen in the bus duct. The debris screen caused a phase to ground fault on the BFN Unit 3 IPB.

The short-term reactor pressure and level responses are shown below.



The BFN response following generator trips were in accordance with expectations.
The above events were determined to be bounded by the transient event analysis (Generator Load Rejection with and without Bypass) as described in UFSAR Section 14.5.2 and UFSAR Appendix N.

Plant staff familiarization with facility operation and EOPs

The EPU will not change any plant operations or EOP actions associated with a generator load reject/ turbine trip transient. Since the dome pressure does not change, no SRV set point changes are required. Operation of the Pressure Control System has not changed for EPU. EPU Power Ascension Testing (Test No. EPU-22) will ensure proper reactor pressure control.

Margin reduction in safety analysis for Anticipated Operational Occurrences (AOOs)

Pressurization transients were analyzed using the approved transient analysis methodology documented in FUSAR Section 2.8.5.2. A generator load rejection is not a limiting event for pressure and does not result in a reduction to the margin of safety. The MSIV closure event (FUSAR Section 2.8.4.2) is more limiting than the Turbine Trip event with respect to reactor overpressure. The EPU evaluations show a 30 to 40 psi difference between these two events. In addition, an evaluation of the MSIV closure event is performed with each reload analysis. The MSIV closure transient analysis was previously discussed above.

EPU Power Ascension Testing

Turbine trip/generator load rejection tests from approximately 100% core power during EPU power ascension testing are not required for BFN. The plant response at EPU conditions is expected to be similar to those documented in the initial startup testing program and those experienced during plant operation. The transient analysis performed for the BFN EPU demonstrates that all safety criteria are met, and that EPU does not cause this event to become limiting. Deliberately causing a load reject and subsequent scram from 100 percent power will result in an undesirable transient cycle on the primary system that can cause undesirable effects on equipment and grid stability. The load rejection transient provides no benefit to safety equipment. Additional turbine trip/load reject testing would result in plant response that has been previously observed, and the test would not provide any new insights into SSCs performance.

Reactor pressure remains constant and the SRV set points do not change for EPU. The steam flow is increased for EPU and there are no changes to the steam bypass capacity. Because of these changes, an increase in peak reactor pressure will occur. As a result, the EPU analysis predicts that the SRVs will lift in the relief mode during a Turbine Trip and Generator Load Rejection Event. Opening SRVs is consistent with OLTP observations for this event.

Since initial startup testing, the original analog Feedwater Control System was replaced with a digital Feedwater Control System that has a Scram Response Logic. This logic reduces overfilling the reactor vessel following a reactor scram with a low reactor water level. When activated, the logic selects one of the three Feedwater Pumps for automatic level control. For the selected pump, the logic limits their speed to approximately 4100 rpm (normally set at 5600 rpm). For the remaining pumps, the logic sets their speed at 600 rpm (normally set to 5600 rpm), or 0% output. These functions limit the transient level overshoot following a scram from a high power level. As discussed in PUSAR Section, 2.4.1.2.3, the Feedwater Control System at BFN meets all CLTR dispositions because the FW flow instrumentation and steam flow instrumentation will be recalibrated or replaced (as necessary) prior to EPU implementation as shown in Table 2.4-2.

The modifications to the feedwater and condensate system were evaluated for this event. Operational level control strategies as well as design requirements ensure that a level 8 trip is avoided. This is consistent with original startup testing (Test No. 27) acceptance criteria. EPU Power Ascension Testing (Test No. EPU-23) will ensure that level control system and reactor feed pump response is consistent with the original start up test requirements. EPU Power Ascension Testing (Test No. EPU-22) will ensure proper steam pressure control. Compliance with these requirements will ensure the level 8 trip will be avoided. The feedwater control system and pressure control system response testing outlined in Table 46-1, Comparison of BFN Initial Startup Testing and Planned Testing, and Table 46-2, Planned EPU Power Ascension Testing, will verify the required system response to address the EPU modification and system changes.

Conclusion

The operating history of BFN demonstrates previous turbine trip/load reject transient events from full power (OLTP and CLTP) are within expected peak limiting values. Based on past transient testing, past analyses and the evaluation of test or actual event results, the effects of a trip from 100 percent EPU power can be analytically determined. No new design functions necessitating modifications and large transient testing validation are required of safety related systems for the EPU. No physical modification or setpoint changes were made to the pilot operated SRVs. The EPU turbine trip transient analysis indicates adequate margin remains to the code limits. The increase in steam flow and its impact on bypass capacity is not significant with regard to the reactor pressure transient response. The changes to the feedwater system do not adversely change the feedwater level control response and are predicted to improve the response.

In view of the above, transient mitigation capability is demonstrated by post modification testing and Technical Specification required testing. In addition, the limiting transient analyses are included as part of the cycle specific reload licensing analysis. From a safety significance standpoint, turbine trip/load reject testing cannot be justified. The transient cycle on the primary plant is undesirable and the potential benefits from such a cycle are not safety-significant. The response of the reactor and its control systems following trips of the turbine and generator has been demonstrated by numerous plant events and shown by EPU analysis to be acceptable. Therefore, this test is satisfied without requiring actual plant transient testing and this test is not warranted.

5.4.3. Other Large Transient Tests

The remaining large transient tests performed at ≥ 80% OLTP during the original power ascension of BFN Units 1, 2, and 3 shown in Table 4-1 were reviewed for applicability to the EPU power ascension test plan. This section compares original start up test data, actual past plant events (if available), CLTR recommendations, and the EPU analysis performed to justify eliminating these large transient tests for the EPU.

Feedwater Pump Trip

Initial Startup Test Objectives and Results

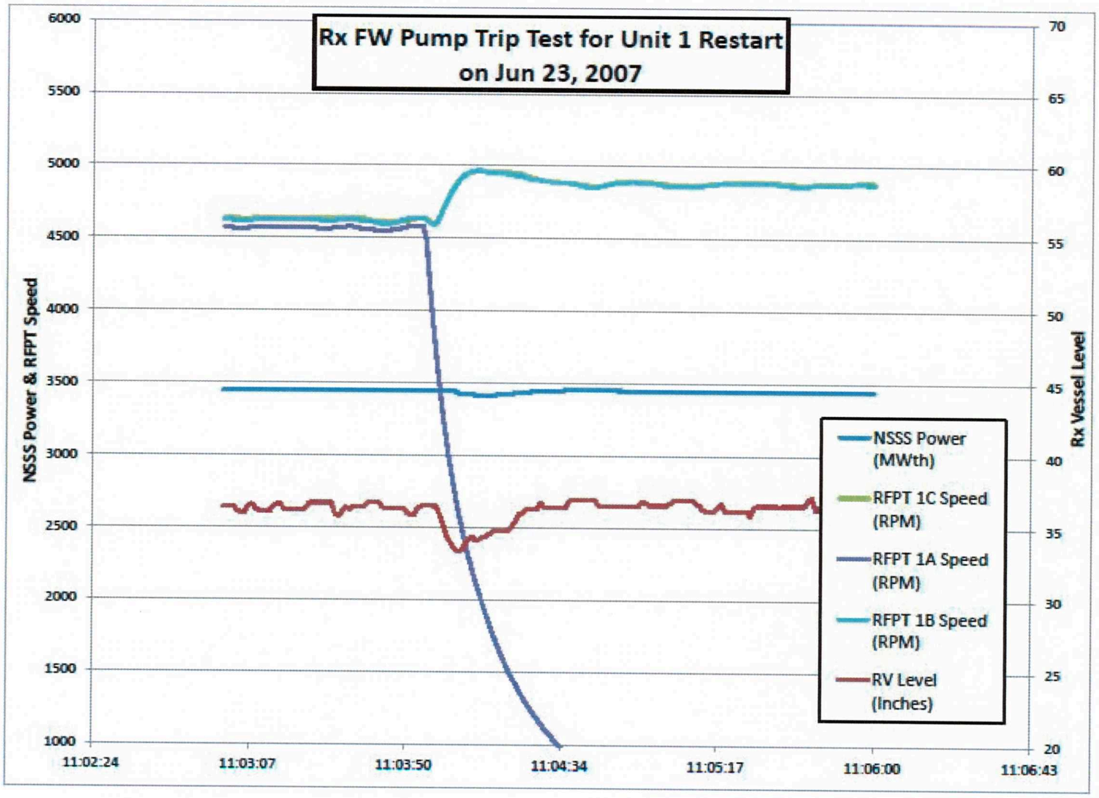
The original startup testing included trips of one of the three operating feedwater pumps at 100% power. The purpose of this test is to demonstrate the capability of the recirculation flow runback feature to prevent a low water scram following the trip of one feedwater pump. A description of the test and acceptance criteria is contained in the BFN UFSAR Section 13.5.2 and 13.5.3 (Test No. 23).

The feedwater and recirculation systems responded satisfactorily to the feedwater pump trip, and all criteria were satisfied. The minimum reactor water level reached was well above the scram setpoint.

Unit 1 Restart Test Program and Results

For the Unit 1 restart, the NRC required Condensate and Feedwater Pump Testing. As part of the power uprate program, the condensate pumps, the condensate booster pumps, and the feedwater pumps were either modified or replaced (See Attachment 47 for a description of changes). These upgrades allowed two (2) of three (3) pumps to supply 100 percent flow rate. The test (1-TI-537) tripped a condensate booster pump, a condensate pump, and a main feedwater pump on an individual basis (i.e., one pump at a time). Following each pump trip, correct transient response was confirmed.

The Feedwater Pump Trip transient test was performed from approximately 3448 (104.7 percent OLTP). All acceptance criteria were satisfied. The short-term Reactor Vessel Level, Feedwater Pump Speeds, and NSSS Power responses are shown below for the RFPT trip test. The graph shows acceleration of the running Feedwater Pumps and a rapid recovery in Reactor Vessel Level. NSSS Power is relatively constant showing that the Recirculation Pumps did not runback and reduce power.

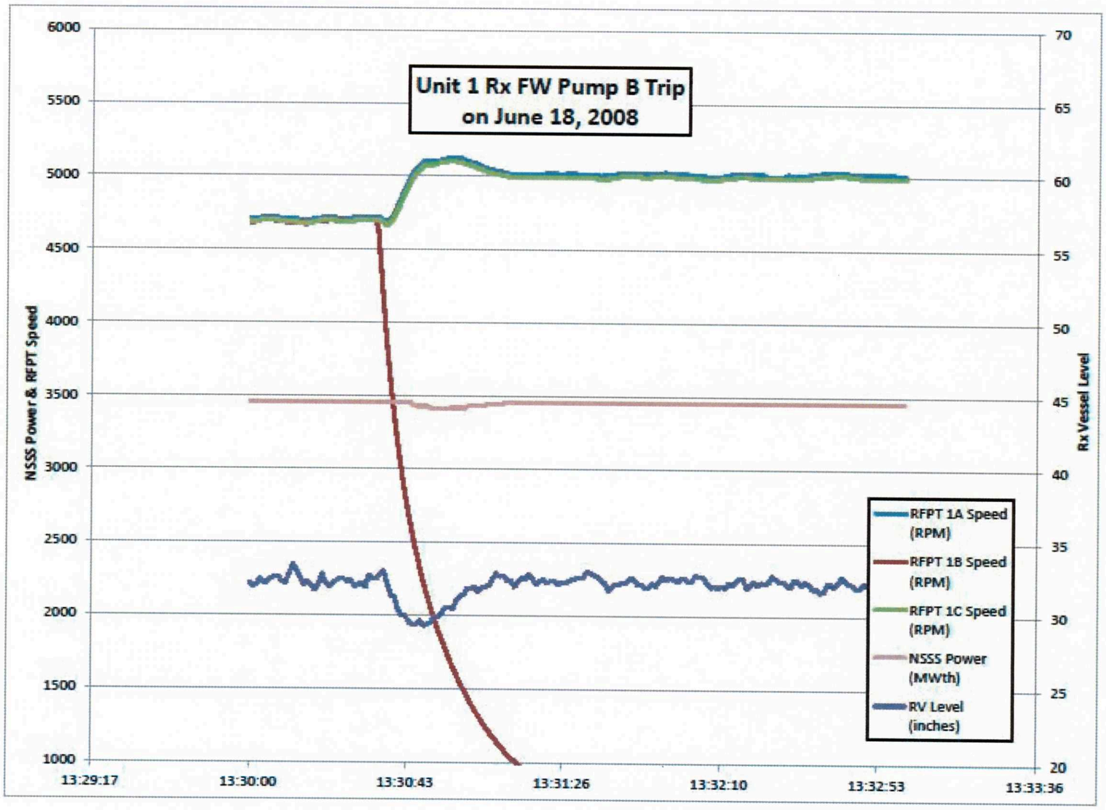


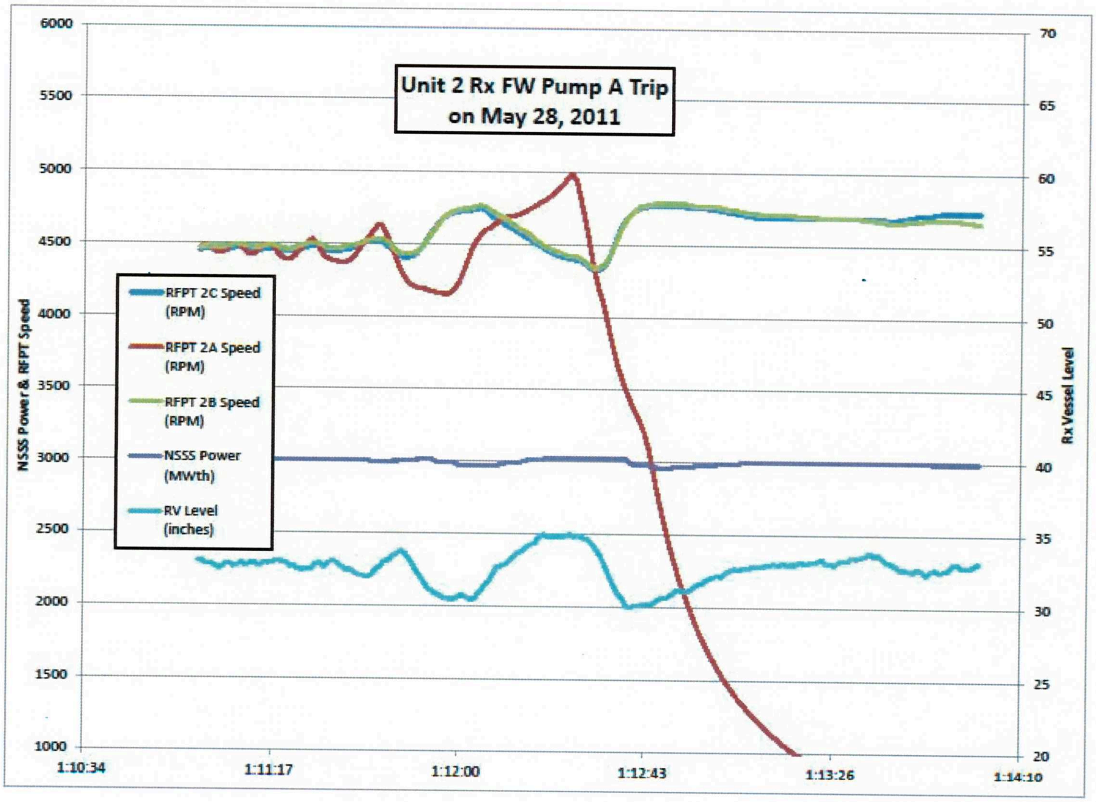
BFN Operating Experience

Since increasing power to the CLTP and upgrading the feedwater and condensate systems BFN has had two feedwater pump trips. Unit 1 experience a feedwater pump trip on June 18, 2008 from approximately 100% CLTP, and Unit 2 experienced a feedwater pump trip on May 28, 2011 from approximately 87% CLTP. Neither transient initiated the recirculation pump runback. The upgraded feedwater pumps were able to supply required flow to prevent a low-level alarm. Table 5-4 lists key parameters, and the following figures show the reactor level transient and Reactor Feedwater Pump speed response.

Table 5-4 Reactor Feedwater Pump Trips

Unit	Event Date	Approximate Power Level (% CLTP)	Lowest Level (Inches)	Margin to Recirc Runback (Inches)	Margin to Level 3 SCRAM (Inches)
1	06/18/2008	100	29.3	0.3	27.3
2	05/28/2011	87	30.0	1.0	28.0


Unit 1 Rx FW Pump B Trip
on June 18, 2008
NSSS Power & RFPT Speed
Rx Vessel Level
6000
5500
5000
4500
4000
3500
3000
2500
2000
1500
1000
70
65
60
55
50
45
40
35
30
25
20
13:29:17
13:30:00
13:30:43
13:31:26
13:32:10
13:32:53
13:33:36
RFPT 1A Speed (RPM)
RFPT 1B Speed (RPM)
RFPT 1C Speed (RPM)
NSSS Power (MWth)
RV Level (inches)



Vendor Topical Report

ELTR1 states that the single feedwater pump trip response is not affected by a power uprate. The response is affected by the flow control line, and uprate utilizes the same MELLLA flow control that BFN uses at CLTP.

Analysis

The single feedwater pump trip transient was analyzed at EPU operating conditions to determine if the level 3 trip setpoint is avoided on a loss of one FW pump. With the planned EPU configuration, a feedwater pump trip at EPU operating conditions would not result in a Scram on level 3 since the minimum water level is 10 inches above the level 3 setpoint (FUSAR Section 2.8.5.2.3.2).

The BFN EPU analysis also included a total loss of feedwater event, which demonstrates the ability to maintain reactor level above the top of active fuel. This analysis relied only on the RCIC system to restore reactor water level. Slightly more time is required for the automatic systems to restore level due to the additional decay heat from EPU. Analysis shows that level is maintained 78 inches above the top of active fuel at EPU condition. After water level is restored, the operator manually controls water level, reduces reactor pressure, and initiates RHR shutdown cooling. This sequence of events does not change or require any new operator actions, or shorten any operator response times. Therefore, operator actions for a Loss of Feedwater (LOFW) transient do not change for EPU.

Conclusion

TVA has reviewed BFN's initial startup testing, restart testing, BFN and industry operating experience, analysis, and ELTR guidance. The test results and recent BFN operating experience, at increased power levels, demonstrate adequate plant response to this event and further testing is not considered necessary. The plant response to a reactor feedwater pump trip has been demonstrated at current licensed power. Supporting analysis included as part of EPU also demonstrates continued safe operation and expected plant response to the loss of feedwater event after EPU. EPU Power Ascension Testing (Test No. EPU-23) will ensure that level control system and reactor feed pump response is consistent with the original start up test requirements. Therefore, the objective of this test is considered satisfied without requiring new or additional transient testing.

Loss of Feedwater Heating (Unit 1 Test only)

Initial Startup Test Objectives and Results

On BFN Unit 1 from 85 percent power, extraction steam heating for a high pressure feedwater string was isolated to demonstrate the plants response to inlet sub-cooling changes on reactor power and pressure.

The transient resulted in approximately a 20 degree reduction in feedwater temperature and less than four percent rise in core power. All core parameters remained within limits.

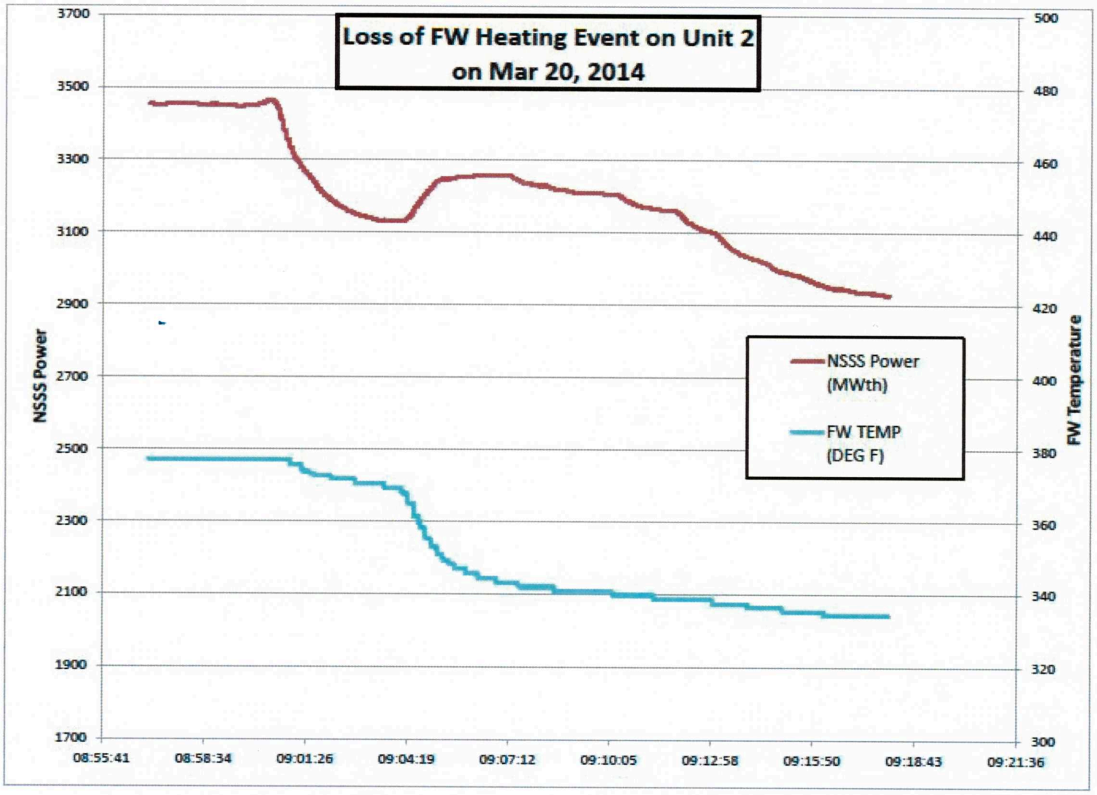
BFN Unit 2 and BFN Unit 3 did not perform the Loss of Feedwater Heater test.

BFN Operating Experience

PER 863920 (BFN Unit 2 Feedwater Heater Extraction Steam Isolation)

On 3/20/2014 at approximately 0900, Unit 2 received a low pressure feedwater heater extraction steam isolation on the 2A string of low pressure heaters (2A3, 2A4, 2A5). Soon after, a level perturbation in the 2A feedwater heater drain system caused the 2A1 and 2A2 high pressure heaters extraction steam to isolate. Unit operator reported 2A5 low pressure heater high level drain valve full open with no indication from the 2A5 normal level drain valve. Unit 2 then entered 2-AOI-6-1C, and lowered reactor power per the immediate actions of the AOI. Due to the combinations of heaters out of service, per 2-OI-6, reactor power was lowered to maintain below a maximum of 952 MWe.

The short-term NSSS Power and final FW Temperature responses are shown below.



Vendor Topical Report

The CLTR established a standard set of tests for the initial power ascension to CPPU. A Loss of Feedwater Heating is not included in this set of tests. PUSAR Section 2.12.1.1 successfully confirms the generic assessment; therefore, the vendor does not require a Loss of Feedwater Heating test.

Analysis

The loss of feedwater heating event analysis supports an assumed 100°F decrease in the feedwater temperature. The result is an increase in core inlet subcooling, which reduces voids, thereby increasing core power and shifting axial power distribution toward the bottom of the core. As a result, of the axial power shift and increased core power, voids begin to build up in the bottom region of the core, acting as negative feedback to the increased subcooling effect. The negative feedback moderates the core power increase. Although there is a substantial increase in core thermal power during the event, the increase in steam flow is much less because a large part of the added power is used to overcome the increase in inlet subcooling. The increase in steam production is accommodated by the pressure control system via the TCVs or the turbine bypass valves, so no pressurization occurs. A cycle-specific analysis is performed.

Conclusion

The Loss of Feedwater Heater test performed during BFN Unit 1 initial startup testing and BFN operating experience demonstrate adequate plant response to this transient. The transient can be caused by an equipment failure or an operator error that causes isolation of one or more feedwater heaters. Plant-specific transient analyses from previous cycles demonstrate acceptable response relative to fuel thermal limits; i.e., minimum critical power ratio (MCPR) and fuel overpower. Loss of Feedwater Heating is a fuel thermal margin event, and these events are generically addressed in the CLTR, Section 9.1.1. The AOO events that determine the operating limit MCPR do not change significantly due to an increase in reactor power up to 20% above the OLTP (FUSAR Table 2.8-1 through Table 2.8-4). The Loss of Feedwater Heater transient was reanalyzed for EPU and fuel thermal limits were acceptable (FUSAR 2.8.5.1). The thermal margin event analysis at Browns Ferry is confirmed to be consistent with the generic description provided in the CLTR. A reduction in electrical output will be required to perform this test, and marginal knowledge to plant operation will be gained from test performance.

Based on plant data and the analytical results, loss of feedwater heating testing will not be conducted as part of EPU power ascension.

Recirculation Pump Trips

Initial Startup Test Objectives and Results

The objective relating to recirculation pump trips is to evaluate the recirculation flow, power and level transients following trips of one or both recirculation pumps. A description of the test and acceptance criteria is contained in the BFN UFSAR Section 13.5.2 and 13.5.3 (Test No. 30)

Single pump trips and double pumps trip were performed at various power levels up to approximately 100 percent power. Single pump trips were initiated by opening the motor generator (MG) field breaker. Double pump trips were initiated by tripping the MG set drive motors. Plant systems responded satisfactorily following each recirculation pump trip, core parameters remained well within limits, and all acceptance criteria were satisfied.

BFN Operating Experience

A new flow control system and Variable Frequency Drives (VFDs) have replaced the original MG flow equipment on BFN Units 1, 2, and 3. After each refueling outage, testing is performed to ensure proper control system operation. The testing scope is based on maintenance, previous testing, and plant conditions.

Since initial startup several recirculation pump trips have occurred. Examples of three such transients which occurred at or near 100 percent CLTP (105% OLTP) are provided below.

PER 62966 (BFN Unit 3 VFD 3B Removed from Service)

On June 9, 2004, Variable Frequency Drive 3B coolant system suffered a major leak necessitating the rapid removal of the drive from service. Prior to the event, BFN Unit 3 was operating at approximately 3458 MWth (105 percent OLTP). After the sudden reduction in core flow, plant control systems recovered reactor vessel level and pressure. Operations entered Technical Specification LCO 3.4.1, Recirculation Loops Operating.

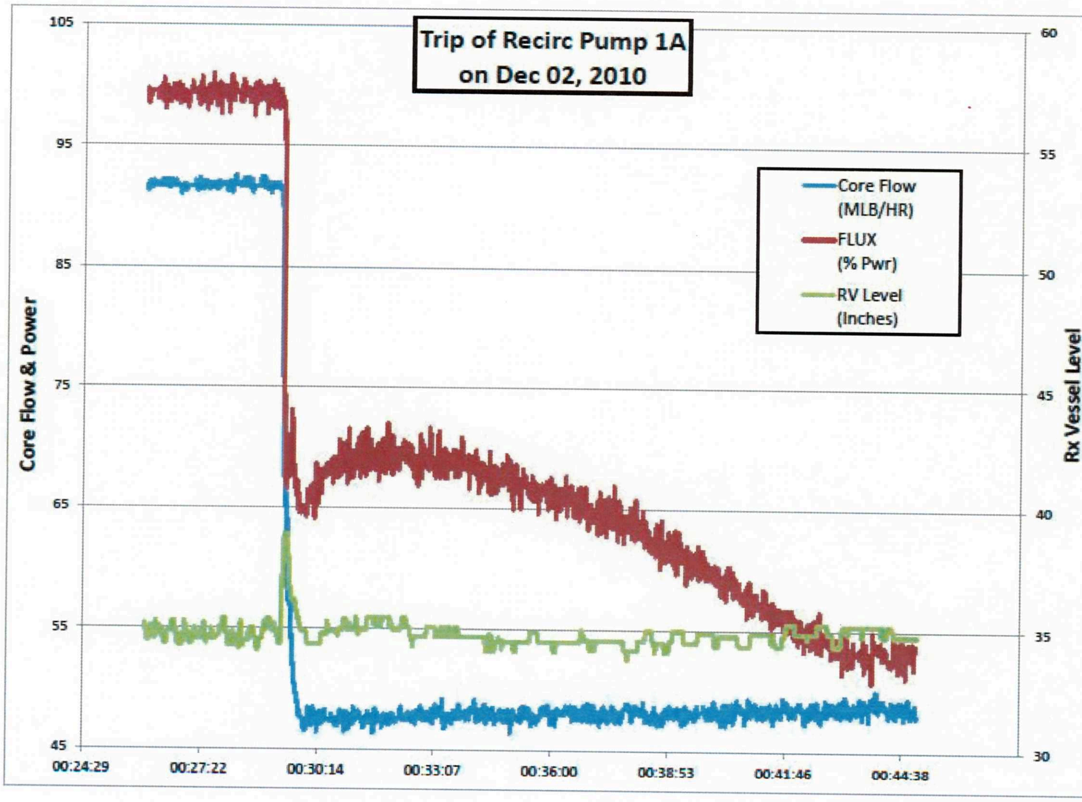
PER 123774 (BFN Unit 2 Recirculation Pump 2B Trip)

On April 23, 2007, Recirculation Pump 2B tripped because of an electrical ground and associated protective relay action. Prior to the event, BFN Unit 2 was operating at 3454 MWth (104.9 percent OLTP) on a 98.6% rod line. After the sudden reduction in core, plant control systems recovered reactor vessel and pressure. Operations entered Technical Specification LCO 3.4.1, Recirculation Loops Operating.

PER 291293 (BFN Unit 1 VFD 1A Unexpected Shutdown)

On December 2, 2010, VFD 1A unexpectedly shutdown due to an equipment controller failure. Prior to the event, BFN Unit 1 was operating at 3452 MWth (104.8 percent OLTP) on a 106.6% rod line. After the sudden reduction in core flow, plant control systems recovered reactor vessel level and pressure. Operations entered Technical Specification LCO 3.4.1, Recirculation Loops Operating.

The short-term core flow, power, and reactor vessel responses are shown below for the BFN Unit 1 event.



In each of the above transients, the reactor did not scram and all plant systems responded as expected. For a CPPU, the recirculation flow rate increases slightly over current operating conditions to provide 100% core flow.

Vendor Topical Report

ELTR 1 does not recommend recirculation pump trip testing because previous tests have shown the plant has large margins to the high level trip set point. In addition, ELTR1 states this level trip margin is not expected to significantly decrease at uprated power level.

Conclusion

TVA has reviewed the initial startup testing, BFN operating experience, analysis, and ELTR guidance. The plant response to a recirculation pump trips has been demonstrated at current licensed power. Supporting analysis included as part of EPU also demonstrates continued safe operation and expected plant response to the recirculation pump trip event after EPU. Due to the small increase in recirculation rate, the plant response is not significantly affected and no additional testing is required. A reduction in electrical output will be required to perform this test, and marginal knowledge to plant operators will be gained from the test performance.

Therefore, the objective of this test is considered satisfied without requiring new or additional transient testing.

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing															
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
1 (See Test No.92 for Unit 1 steam dryer)	**CHEMICAL AND RADIOCHEMICAL** This test verified plant water quality, radiochemistry, and the proper functioning of water purification equipment. The Unit 2&3 test also verified steam separator – dryer performance.	U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	EPU-1A EPU-1B for steam separator – dryer	Test will be performed. See Table 46-2 for description.			X	X	X	X
2	**RADIATION MEASUREMENTS** This test verified proper radiation postings in the plant. 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, 75 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.	U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	EPU-2	Test will be performed. See Table 46-2 for description.			X	X	X	X
3	**FUEL LOADING** This test loaded fuel safely and efficiently to the full core size.	U1 U2 U3						NO	No new fuel loading methods or techniques are needed for the EPU core load. Fuel load will be performed utilizing BFN's plant procedures and cycle reload analysis.						

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing															
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
4	**FULL CORE SHUTDOWN MARGIN** This test demonstrated that the reactor can be subcritical throughout the first fuel cycle with any single control rod fully withdrawn.	U1 U2 U3						NO	Shutdown margin (SDM) is determined every core reload in accordance with BFN's plant surveillance procedures. Compliance with plant Technical Specifications ensures that SDM is within the limits provided in the COLR.						
5	**CONTROL ROD DRIVE** This test (a) demonstrated that the Control Rod Drive (CRD) system operated properly over the full range of primary coolant temperatures and pressures from ambient to operating, and (b) determined the initial operating characteristics of the entire CRD system.	U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 -- --	-- -- U3	U1 U2 U3	Normal surveillance tests will be performed.	CRD system was evaluated in the CLTR (section 2.5). The CLTR states that (1) scram time performance relative to current plant operation is bounding, and (2) CRD positioning is not affected by CPPU. The PUSAR (section 2.8.4.1) confirms the CLTR's generic assessment. Compliance with plant Technical Specifications ensures proper CRD system operation.	X					X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing																
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)						
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
6	**SRM PERFORMANCE AND CONTROL ROD SEQUENCE** This test demonstrated 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. The effect of typical rod movements on reactor power was determined.	U1 U2 U3	U1 U2 U3	U1 U2 U3	-- -- --			NO	SRM instruments are calibrated and tested per plant surveillance requirements. PUSAR (section 2.4.1.1.1) confirms the CLTR disposition associated with SRM instruments. By compliance with normal plant surveillance procedures, the IRMs may be adjusted to achieve adequate overlap with SRMs and APRMs. This overlap adjustment is performed with test EPU-10. Control rod sequences are developed in accordance with approved procedures. PUSAR (section 2.4.1.3) confirms CLTR disposition that CPPU has insignificant impact on rod control.						

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing

UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
9 (See Test No. 39 for Unit 1)	**WATER LEVEL MEASUREMENT** This test verified the calibration and agreement of narrow range and wide range level instrumentation at various plant conditions that may impact reference leg head (temperature, vessel pressure and flow).		-- U2 U3	-- U2 U3			-- U2 U3	Monitor per EPU-101, and calibrate instruments using existing procedures.	Monitoring will be performed. The reactor bottom head and the reactor water level instrumentation leg temperatures are unaffected by the EPU. As discussed in PUSAR (Section 2.7.5), the CPPU does not change reactor pressure (temperature) and small drywell temperature changes due to higher feed-water temperature are negligible with respect to water level measurement. Monitoring procedure will confirm instrument performance at uprate conditions.	X	X	X	X	X	X
10	**IRM CALIBRATION** This test adjusted the Intermediate Range Monitor System to obtain an optimum overlap with the SRM and APRM Systems.	U1 U2 U3	U1 U2 U3	U1 U2 U3				EPU-10	Test will be performed. See Table 46-2 for description.	X					
11	**LPRM CALIBRATION** This test calibrated the Local Power Range Monitor System to make the LPRM readings proportional to the neutron flux in the narrow-narrow water gap at the chamber elevation.			U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	Normal surveillance tests will be performed.	The purpose of this test is to calibrate the LPRMs to read proportional to the neutron flux in the core. As discussed in PUSAR (section	X					X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing

UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
									2.4.1.1) the increase in neutron flux is within the design of the LPRM system. Modification to the LPRM system is not required for EPU. Therefore, specific EPU testing is not required. Compliance with plant Technical Specifications ensures proper LPRM operation.						
12	**APRM CALIBRATION** This test (1) provided a method to calibrate Average Power Range Monitor Channels prior to the first heatup, and (2) directed the calibration of APRM channels to read percent of core thermal power after an accurate heat balance determined core thermal power.		U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	EPU-12	Test will be performed. See Table 46-2 for description.	X	X	X	X	X	X
13	**PROCESS COMPUTER** This test verified that the computer indicated correct values of sensed process variables and that the results of calculations were correct.	U1 U2 U3	U1 U2 U3	U1 -- U3	-- U2 U3		U1 U2 U3	Verify the capability of the process computer to monitor plant conditions and to evaluate core performance parameters per plant procedures.	Process values are verified during instrument calibrations. Operation of the process computer is not affected by EPU, and plant procedures maintain the validity of computer calculation. Data installation and calculation are verified during startup from each refueling outage	X					X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing

UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
									use plant test procedure discussed in UFSAR Section 13.10.						
14	**RCIC SYSTEM** This test verified the ability of the RCIC system to provide the required flow rate at various turbine steam supply and pump discharge pressures and to start from cold standby conditions.		U1 U2 U3	U1 U2 U3				NO	PUSAR (Section 2.8.4.3) confirms CLTR (Section 3.9), that CPPU does not produce any changes to RCIC system. Pressures, temperatures, flow rates, and timing requirements are unchanged. Compliance with plant Technical Specifications ensures proper RCIC system operation.						
15	**HPCI SYSTEM** This test verified the ability of the HPCI system to provide the required flow rate at various turbine steam supply and pump discharge pressures and to start from cold standby conditions.		U1 U2 U3		U1 U2 U3			NO	PUSAR (Section 2.8.5.2.6.1) confirms CLTR (Section 4.2), that CPPU does not produce any changes to HPCI system. Pressures, temperatures, and flow rates, requirements are unchanged. Compliance with plant Technical Specifications ensures proper HPCI system operation.						

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing

UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
16	**SELECTED PROCESS TEMPERATURES** This test established the proper setting of the low speed limiter for recirculation pumps to avoid coolant temperature stratification in the reactor pressure vessel bottom head region.		U1 U2 U3	U1 U2 U3	U1 U2 U3			NO	PUSAR (Section 2.8.4.6) discusses the impact of a CPPU on the Recirculation System. As discussed, flow interlocks in terms of absolute values are not changed by EPU.						
17	**SYSTEM EXPANSION** This test verified that the reactor drywell piping system is free and unrestrained in regard to thermal expansion and that suspension components were functioning in the specified manner.	U1 U2 U3	U1 U2 U3	U1 U2 U3				NO	As discussed in PUSAR (Section 2.2.2.2), the piping systems' thermal expansion is unaffected by EPU (negligible changes in system temperatures due to EPU) except for feedwater piping. The effect of the rated feedwater temperature increase on pipe movement during cold to hot cycling is negligible for testing purposes. Also, there are no physical changes to drywell piping systems.						
18	**CORE POWER DISTRIBUTION** This test: (1) confirmed the reproducibility of the TIP system readings, (2) determined the core power distribution in three dimensions, and (3) determined core power symmetry.			U1 U2 U3	U1 -- U3	U1 U2 U3	U1 U2 U3	EPU-18	Test will be performed. See Table 46-2 for description.	X Note 3		X Note 3			X Note 3

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing															
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
19	**CORE PERFORMANCE** This test evaluated the core and thermal hydraulic performance to ensure parameters are within limits.			U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	EPU-19	Test will be performed. See Table 46-2 for description.	X	X	X	X	X	X
20	**[U1] ELECTRICAL OUTPUT AND PRELIMINARY HEAT RATE TEST** This test is to demonstrate that the guaranteed gross electrical output requirements are satisfied without exceeding the reactor power level warranty and to determine a preliminary net plant heat rate value. **[U2] ELECTRICAL OUTPUT AND HEAT RATE TEST** This test is to demonstrate that the plant net electrical output and net heat rate requirements are satisfied. **[U3] STEAM PRODUCTION** This test demonstrated that the Nuclear Steam Supply System provided steam sufficient to satisfy all appropriate warranties.						U1 U2 U3	Baseline turbine thermal cycle performance and plant heat rate will be determined using the existing thermal performance program	No specific warranty testing in support of EPU implementation is required.						X
21	**FLUX RESPONSE TO RODS** This test demonstrated the stability of the power-reactivity feedback loop with increasing reactor power and determined the effect of control rod movement on reactor stability.			U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	Existing plant procedures	As discussed in PUSAR (Section 2.12.1.1), 120% of OLTP power is achieved on the previously licensed MELLLA rod line. EPU adds a region to the power to flow map where thermal-hydraulic stability is not a concern Analytical stability evaluations are core reload dependent and are performed for each			X			X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing															
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
									reload fuel cycle in accordance with approved procedures. The CLTR does not require specific testing. Therefore, existing rod positioning procedures will be utilized to demonstrate reactor stability. Also, reactor core stability monitoring is performed continuously by a dedicated system and by operator oversight.						
22	**PRESSURE REGULATOR** This test (a) determined the optimum settings for the pressure control loop by analyzing the transients induced in the reactor pressure control system by means of the pressure regulators, (b) demonstrated the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator and set the spacing between set points at an appropriate value, and (c) demonstrated smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam used by the turbine.			U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	EPU-22	Test will be performed. See Table 46-2 for description.	X	X	X	X	X	X Note 1
23	**FEEDWATER SYSTEM** This test (a) adjusted the feedwater control system for acceptable reactor water level control, (b) demonstrated stable reactor response to sub-cooling			U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	EPU-23	(a) Test will be performed. See Table 46-2 for description. PUSAR (Section	X	X	X	X	X	X Note 1

UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
	changes [Unit 1 only], (c) demonstrated the capability of the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump.								2.4.1.2.3) discusses the Feedwater Control System. With the implementation of instrument changes shown in Table 2.4.-2, the Feedwater Control System will meet all CLTR dispositions. (b) & (c) Loss of FW Heating and Feedwater pump trip testing are large load transients, and will not be performed (see Section 5).						
24	**BYPASS VALVES** This test (a) demonstrated the ability of the pressure regulator to minimize the reactor pressure disturbance during an abrupt change in steam flow, and (b) demonstrated that a bypass valve can be tested for proper functioning at rated power without causing a high flux scram.			U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	EPU-24	The surveillance test will be performed to determine the maximum power level that the test can be performed without a scram or isolation. See Table 46-2 for description.	X	X	X	X	Based on prior test data Note 2	Based on prior test data Note 2
25	**[U1] MAIN STEAM ISOLATION VALVES** This test (a) functionally checked the main steam line isolation valves (MSIVs) for proper operation at selected power levels, (b) determined reactor transient behavior during and following full closure of one valve, (c) determined isolation valve closure time. **[U2&U3] MAIN STEAM ISOLATION VALVES**		U1 U2 U3		U1 U2 U3	U1 U2 U3	-- U2 U3	Normal surveillance tests (Closure Testing) will be performed per the IST Program. Surveillance test for RPS	The RPS functional surveillance test will be performed on at least one MSIV (most limiting) to determine the maximum power level that the test can be performed without a scram or isolation.	X		X	X	Based on prior test data Note 2	Based on prior test data Note 2

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing																
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)						
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
	This test (a) functionally checked the main steam line isolation valves (MSIVs) for proper operation at selected power levels, (b) determined reactor transient behavior during and following simultaneous full closure of all MSIV's and following full closure of one valve, (c) determined isolation valve closure time, and (d) determined the maximum power at which a single valve closure can be made without scram.							function will be tested at power.	As discussed in PUSAR (Section 2.2.2.2.1.2), MSIV performance is bounded by conclusions of the evaluation in Section 4.7 of ELTR2, and the Browns Ferry MSIVs are acceptable for EPU operation. BFN InService Testing (IST) Program establishes the testing and examination requirement to assess operational readiness of the MSIVs. Per the IST Program, single MSIV closures are not performed at power at BFN. Only partial MSIV stroke tests per surveillance tests are performed to verify associated RPS functions. Fast closure full-stroke time testing of the MSIVs occurs during refueling outage. Simultaneous full closure of all MSIV's at power testing is a large load transient, and						

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing															
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
									will not be performed (see Section 5).						
26	**RELIEF VALVES** This test: (a) verified the proper operation of the primary [steam] relief valves, (b) determined their capacity, and (c) verified proper reseating following operation.		U1 U2 U3	U1 U2 U3	U1 -- --	U1 -- U3		NO	As discussed in FUSAR (Section 2.8.4.2), primary relief valve existing setpoints, and existing capacity meet EPU requirements. No additional testing for EPU is required. Compliance with plant Technical Specifications ensures proper operation of the primary relief valves.						
27	**[U1] TURBINE STOP AND CONTROL VALVE TRIPS** This test (a) determined the response of the reactor system to a turbine stop or control valve trip and (b) evaluated the response of the bypass, relief valve and reactor protection systems. The parameters of particular interest were peak values and the rate of change of both reactor power and reactor steam dome pressure. **[U2&U3] TURBINE TRIP AND GENERATOR LOAD REJECTION** This test demonstrated the response of the reactor and its control systems to protective trips in the turbine and generator. The parameters of peak values and change rates of reactor steam pressure and heat flux were			U1 U2 U3		U1 -- U3	U1 U2 U3	NO	Turbine and generator trip testing is a large load transient, and will not be performed (see Section 5).						

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing

UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
	determined. The ability to perform a load rejection within bypass capacity without a scram was demonstrated.														
29 (See Test No. 32 for Unit 3)	**FLOW CONTROL** This test (a) determined the plant response to changes in the recirculation flow, (b) optimized controllers, and (c) demonstrated the plant load following capability in flow control mode. [Note: Testing performed in conjunction with Test No 32.]			U1 U2 --	U1 U2 --	U1 U2 --	U1 U2 --	Normal cycle startup test per BFN's plant procedures will be performed.	PUSAR (Section 2.8.4.6) discusses the impact of a CPPU on the Recirculation System. As stated the increased voids in the core during normal uprated power operation requires a slight increase in the recirculation drive flow to achieve the same core flow. This results in a small change in the rated recirculation flow rate. This parameter is considered negligible and does not invalidate previous test results. A new flow control system and Variable Frequency Drives (VFDs) have replaced the original flow control and drive equipment. After each refueling outage, testing is performed to ensure proper control system operation. The test scope is based on maintenance, previous testing, and plant conditions.	X		X	X	X	X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing																
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
30	**RECIRCULATION SYSTEM** This test (a) evaluated the recirculation flow and power level transients following trips of one or both of the recirculation pumps, (b) obtained recirculation system performance data, and (c) verified that no recirculation system cavitation occurred on the operable region of the power-flow map.				U1 U2 U3	U1 U2 U3	U1 U2 U3	(a) NO (b) Core flow calibration will be performed per plant procedures (c) NO	(a) Recirculation pump trip testing is a large load transient, and will not be performed (see Section 5). (b) Core flow calibration will be performed per plant procedures (c) EPU does not change the rated flow for the reactor recirculation pumps. Slightly higher RR pump speeds are required to produce the same flow, this parameter is considered negligible and does not invalidate previous test results. As discussed in PUSAR (2.8.4.6), flow interlocks in terms of absolute values are not changed by EPU. Therefore, verification of non-cavitation is not required.			X			X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing															
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
31	**LOSS OF TURBINE-GENERATOR AND OFFSITE POWER** This test verified that the reactor can safely withstand a loss of the turbine-generator and all off-site power, and demonstrated acceptable performance of the station electrical supply system			U1 U2 U3				NO	Turbine-generator trip and loss of offsite power testing is a large load transient, and will not be performed (see Section 5).						
32	**[U1] RECIRCULATION M-G SET SPEED CONTROL** **[U2&3] RECIRCULATION M-G SET SPEED CONTROL AND LOAD FOLLOWING** This test adjusted recirculation control system parameters to obtain optimum speed control performance. (Load following was not performed.)	U1 -- --	U1 -- --	U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	Normal cycle startup test per BFN's plant procedures will be performed.	Same evaluation as Test No.29.	X		X	X	X	X
33	**MAIN TURBINE STOP VALVE SURVEILLANCE TEST** This test demonstrated an acceptable procedure for turbine stop valve testing at a power level as high as possible without producing a reactor scram.			U1 U2 U3	U1 U2 U3	U1 U2 U3	U1 U2 U3	EPU-24	Test will be performed. See Table 46-2 for description.	X	X	X	X	Based on prior test data Note 2	Based on prior test data Note 2
34 (See Test No.90 for Unit 1)	**VIBRATION MEASUREMENTS** This test obtained vibration measurements on various reactor components at various plant conditions to demonstrate the plant integrity to flow induced vibration, and to validate the analytical vibration model.	-- -- U3	-- -- U3	-- -- U3	-- U2 U3	-- U2 U3	-- U2 U3	EPU-100 for key components and piping	Test will be performed. See Table 46-2 and Attachment 45 (Vibration Analysis and Monitoring Program) for a description. Replacement steam dryer monitoring is discussed in Attachment 40.	X	X	X	X	X	X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing															
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
35	**RECIRCULATION AND JET PUMP SYSTEM CALIBRATION** This test performed a complete integrated calibration of the installed jet pump and recirculation system instrumentation. [Note: On Unit 1 at power calibrations were performed under Test No 30.]	U1 U2 U3		-- -- --	-- U2 U3	-- U2 U3	-- U2 U3	Core flow calibration will be performed per plant procedures.	As discussed in PUSAR (2.8.4.6), the physical characteristics of the recirculation system and jet pumps are not changing as a result of CPPU. Therefore, calibration parameters do not change. Existing calibration and plant procedures remain valid, and a specific EPU test is not required.			X			X
36	**EQUALIZER OPEN** This test (a) explored the allowable operating range and performance of the recirculation system under the conditions of one recirculation pump operating and the equalizing valves open to cross-tie the two loops, and (b) provided information for developing operating procedures.				U1 -- --		U1 -- --	NO	BFN does not operate with one recirculation pump and the equalizing valves open. On Units 1 and 3 the equalizer valves have been removed from the recirculation piping, so the loops cannot be cross-tied to provide motive force to all jet pumps from a single pump. On Unit 2, the equalizer valves are still installed but operating restrictions do not permit both valves being open during power operation.						

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing																
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
									Therefore, this test is no longer applicable and is not required for EPU.						
39 (See Test No.9 for Units 2 and 3)	**WATER LEVEL VERIFICATION IN REACTOR VESSEL** This test (a) verified the calibration and agreement of narrow range and wide range level instrumentation at various plant conditions that may impact reference leg head (temperature, vessel pressure and flow), and (b) verified the ability of the feedwater control system to regulate reactor water level at 50% and 100% power		U1 -- --		U1 -- --		U1 -- --	(a) Monitor per EPU-101, and calibrate instruments using existing procedure. (b) EPU-23	(a) Monitoring will be performed. The reactor bottom head and the reactor water level instrumentation leg temperatures are unaffected by the EPU. As discussed in PUSAR (Section 2.7.5), the CPPU does not change reactor pressure (temperature) and small drywell temperature changes due to higher feed-water temperature are negligible with respect to water level measurement. Therefore, specific EPU testing is not required. Monitoring procedure will confirm instrument performance at uprate conditions.. (b) Test will be performed. See Table 46-2 for descriptions.	X	X	X	X	X	X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing

UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
70	**REACTOR WATER CLEANUP SYSTEM** This test demonstrated the operability of the reactor water cleanup system under actual reactor operating temperature and pressure conditions.		U1 U2 U3					NO	As discussed in PUSAR (Section 2.1.7), the CPPU impact to the Reactor Water Cleanup system is negligible. The system is capable of performing its design cleanup function at the EPU without any modifications or changes in operating techniques. RWCU is operated per plant procedures and water chemistry is maintained to meet TRM requirements.						
71	**RESIDUAL HEAT REMOVAL SYSTEM** This test demonstrated the ability of the Residual Heat Removal (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 suppression pool water.		U1 U2 U3					NO	As discussed in PUSAR (Section 2.8.4.4), CPPU may result in slightly longer times to achieve cold shutdown with RHR, but the ability to successfully place this system in different cooling modes is unaffected. RHR is operated per plant procedures. The IST Program ensures proper RHR equipment performance.						

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing																
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
72	**DRYWELL ATMOSPHERE COOLING SYSTEM** This test verified the ability of the Drywell Atmosphere Cooling System to maintain design conditions in the drywell during power operations.		U1 U2 U3		U1 -- --		U1 U2 U3	Monitor per EPU-101	Monitoring will be performed. As discussed in PUSAR (Section 2.7.6.), the slight addition of heat due to the increase in rated feedwater piping temperature in this area has been determined in to be negligible. The system is operated per plant procedures. Parameters will be verified during power ascension as part of the EPU System Performance & Monitoring.	X	X	X	X	X	X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing																
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
73	**COOLING WATER SYSTEMS** This test verified the performance of the Reactor Building Closed Cooling Water (RBCCW) and the raw cooling water systems are adequate with the reactor at rated condition.		-- U2 U3		U1 -- --		-- U2 U3	Monitor per EPU-101	Monitoring will be performed. As discussed in PUSAR (Section 2.5.3.3), there are no significant changes to parameters or system limits due to EPU. There are no EPU modifications or changes in operating techniques. The system is operated per plant procedures. Parameters will be verified during power ascension as part of the EPU System Performance & Monitoring.	X	X	X	X	X	X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing															
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
74	**OFF GAS SYSTEM** This test verified the proper operation of the Off Gas System over its expected operating parameters and determined the performance of the activated carbon absorbers.			-- -- U3	-- -- U3	-- -- U3	-- -- U3	Monitor per EPU-101	Monitoring will be performed. The Offgas system is evaluated in PUSAR (Section 2.5.5.1). This evaluation concluded that all structures, systems and components of the offgas system were acceptable for EPU operation Normal operating plant procedures contain appropriate limits and ensure Technical Specifications requirements are met. No power dependent system functions added or changed. Design operating margins have been reduced as a result of EPU but parameters are within system design limits. . Parameters will be verified during power ascension as part of the EPU System Performance & Monitoring.	X	X	X	X	X	X

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing																
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)						
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
75	**REACTOR SHUTDOWN FROM OUTSIDE THE MAIN CONTROL ROOM** This test demonstrated that the plant is 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 technical specifications is adequate without affecting the safe, continuous operation of the other units, and that the plant emergency operating instruction is adequate.			-- -- U3				NO	Shutdown from Outside the Main Control Room is implemented by AOI-100-2 (Control Room Abandonment). As discussed in PUSAR (Section 2.11.1.1), there are no EPU procedural changes to this AOI. LAR Attachment 47 shows that there are no modifications that would impact shutdown from outside the control room. Therefore, EPU that would not impair the ability to safely scram the reactor and to maintain the plant in Hot Shutdown from outside the control room Therefore, initial testing remains valid, and this test does not need to be performed.						

Table 46-1 Comparison of BFN Initial Startup Testing and Planned EPU Testing																
UFSAR 13.5.2 and 13.5.3 Test No	Original Test Description	Initial Test Power Level % of OLTP (3293 MWth)						Test Planned for CPPU	Evaluation and Justification	Planned CPPU Test Power Level % of CLTP (3458 MWth)					
		Cold	Heatup	25	50	75	100			≤ 90 (3112)	95 (3285)	100 (3458)	104.8 (3623)	109.5 (3788)	EPU (3952)
90 (See Test No.34 for Units 2 and 3)	**VIBRATION MEASUREMENTS** This test obtained vibration measurements on various reactor components at various plant conditions to demonstrate the plant integrity to flow induced vibration, and to validate the analytical vibration model.	U1 -- --	-- -- --	-- -- --	U1 -- --	U1 -- --	U1 -- --	EPU-100 for key components and piping	Test will be performed. See Table 46-2 and Attachment 45 (Vibration Analysis and Monitoring Program) for a description. Replacement steam dryer monitoring is discussed in Attachment 40.	X	X	X	X	X	X
92	**STEAM SEPARATOR – DRYER** This test determined the carryunder and carryover characteristics of the steam separator-dryer. (Steam separator – dryer performance was included in Test No. 1 for Units 2 and 3)		-- -- --	-- -- --	U1 -- --	U1 -- --	U1 -- --	EPU-1B	Test will be performed. See Table 46-2 for description.		X	X	X	X	X

Notes:

1. Planned tests EPU-22 and EPU-23 are operational transients. The last test performance is the last condition before 100% EPU is achieved. The intention is to perform this testing at a power level so that any excursions are less that than 3952 MWth.
2. Surveillance testing of the turbine valves will always be current. Testing disturbs steam flows and possibly vessel dome pressure. Testing determines the maximum power at surveillance testing can be performed without a scram or isolation. Test performance at higher power levels is based on the projected margins to a scram or isolation.
3. Planned test EPU-18 is a normal refueling outage functional test that is being performed at an additional power level (CLTP). The CLTR does not discuss or require performance of this test. Based a review of testing described in UFSAR Section 13.5, the FUSAR (Section 2.12.1) concludes that the use of ATRIUM-10 and ATRIUM 10XM fuel will not impose any additional testing requirements for EPU operation. The additional performance at 100 % CLTP is based on discussion with BFN Reactor Engineering and TVA BWR Fuel Engineering.

   In addition, seven BWR EPU Test Plans were reviewed. Of the seven plants, two plants did not discuss the Core Power Distribution Test in their test plans, three plants did not perform this test for EPU, one plant performed the test only at full EPU power level for EPU, and one plant performed the Core Power Distribution Test twice for EPU, once at less than 90% CLTP and another at full EPU power level.

   Multiple test performances at less than 90% CLTP are not planned for BFN's EPU. This is based on (1) AREVA document 51-9114888-000, Plant Startup Testing Requirements for Power Distribution Uncertainty Verification (based on methodology has been approved by the NRC), and (2) the large thermal margins that exist at low power.

Table 46- 2 Planned EPU Power Ascension Testing

(*) Test Number	Title	Test Description
EPU-1A	**CHEMICAL AND RADIOCHEMICAL**	Samples will be taken and measurements made at selected EPU power levels to determine: 1) the chemical and radiochemical quality of reactor water and reactor feedwater and 2) gaseous activities. Testing will utilize BFN's plant procedures.
EPU-1B	**STEAM DRYER/ SEPARATOR – PERFORMANCE**	Samples will be taken and measurements made at selected EPU power levels to determine steam dryer – separator performance (i.e., moisture carryover). For this test, main steam moisture content is considered equivalent to the steam dryer/separator moisture carryover. The data (and associated trend) will be used as a secondary means to indicate any potential steam dryer issues. Results will also be reviewed to ensure moisture carryover remains below GE analyzed limits. Sampling and analysis will be performed in accordance with BFN station procedures.
EPU-2	**RADIATION MEASUREMENTS**	At selected EPU power levels, gamma dose rate measurements and, where appropriate, neutron dose rate measurements will be made at specific limiting locations throughout the plant to assess the impact of the uprate on actual plant area dose rates. Design documents and UFSAR radiation zones will be updated as necessary. Monitoring will utilize BFN's plant procedures.
EPU-10	**IRM PERFORMANCE**	Each IRM channel reading will be adjusted to be consistent with core thermal power, referenced to EPU level. Test IRM scram and rod block setting and assure IRM overlap with the APRM. The existing plant surveillance procedures, which assure compliance with the Technical Specification limits, will be utilized to satisfy this requirement.
EPU-12	**APRM CALIBRATION**	Each APRM channel reading will be adjusted to be consistent with the core thermal power, referenced to the EPU level, as determined from the heat balance. Assure that the APRM flow-biased scram and rod block setpoints are consistent with EPU operation. Confirm all APRM trips and alarms prior to entering the EPU operating domain. Calibration will be performed in accordance with BFN surveillance procedures.

(*) The listed Test Number is for ease of referencing test descriptions between Tables 46-1 and 46-2. Actual test numbers will be assigned in accordance with TVA's administrative procedures.

Table 46- 2 Planned EPU Power Ascension Testing

(*) Test Number	Title	Test Description
EPU-18	**CORE POWER DISTRIBUTION**	Traversing Incore Probe (TIP) data will be taken to confirm the symmetry of the TIP system readings and valid data is analyzed to determine core power distribution, and check the core power symmetry. Testing will be performed in accordance with existing BFN refueling test procedure.
EPU-19	**CORE PERFORMANCE**	At steady-state conditions, routine measurements of reactor parameters are taken near 90%, 95% and 100% of CLTP along a constant flow control line to be used to increase to maximum EPU power. Core thermal power and core performance parameters are calculated using accepted methods to ensure current licensed and operational practice are maintained. Power increase is along this constant rod pattern line in incremental steps of 5% or less to ensure a careful, monitored approach to maximum EPU power. Measured reactor parameters and calculated core performance parameters are utilized to project those values at the next power level step. Each step's projected values will satisfactorily confirm the actual values before advancing to the next step and the final increase to maximum EPU power.
EPU-22	**PRESSURE REGULATOR**	<u>Pressure Regulator Tuning and Testing</u> Before startup, the pressure control system will be calibrated. The Main Steam Pressure Control System is tested under the following conditions: a) Prior to Main Turbine startup with bypass valves controlling system pressure b) With Main Turbine on-line, at various power levels, with the load limit set high enough that the entire transient is handled by the control valves, c) With Main Turbine on-line, at various power levels, with the load limit set low enough that the steam is released through the bypass system during some part of the transient. The pressure control system response to a pressure setpoint change is accomplished by first making a 3 psi down setpoint change which is followed by a 3 psi up setpoint change after conditions stabilize. Following the 3 psi pressure change, the same testing is performed for a 6 psi change. When testing is completed in one pressure control mode, the other pressure mode is selected and the pressure step change testing is repeated.

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Table 46- 2 Planned EPU Power Ascension Testing

(*) Test Number	Title	Test Description
		With then Main Turbine offline, the "takeover" capability of the Bypass Control System is demonstrated by simulated failure of the Main Steam Pressure Control System in both the Reactor Pressure Control and Header Pressure Control modes. Incremental Regulation (ratio of % change in control signal to % change in steam flow) Steady state parameters of total steam flow and pressure controller output will be collected at ≤ 2.5% EPU increments from 90% CLTP to 100% EPU. Turbine First Stage Scram Bypass Setpoint Validation Parameters of turbine first stage pressure and reactor thermal power are collected to validate the scram bypass permissive setpoint
EPU-23	**FEEDWATER SYSTEM**	Feedwater Control System Testing The feedwater control system response to reactor water level setpoint changes will be evaluated in the indicated control mode (i.e., three element, single element). At each test condition, level setpoint change testing is performed by first making an up setpoint value change, which effects the level setpoint change desired, followed by a down setpoint value change of the same value, after conditions stabilize, in accordance with the following setpoint change sequence. **Sequence** – **Level Setpoint Change** 1 – Step up 2 inches 2 – Step down 2 inches 3 – Step up 3 inches 4 – Step down 3 inches 5 – Step up 4 (±1 inches) 6 – Step down 4 (±1 inches) The 2 and 3 inch level setpoint steps are informational and recommended to demonstrate the feedwater control system response prior to performing the formal level setpoint steps (i.e., 4 inches). The results from the informational level setpoint steps are utilized to anticipate the responses to the formal demonstration test steps, so that effects on the reactor may be anticipated (i.e., power increases, level alarms). The normal feedwater control system mode is three element control, with single element control only being

(*) The listed Test Number is for ease of referencing test descriptions between Tables 46-1 and 46-2. Actual test numbers will be assigned in accordance with TVA's administrative procedures.

Table 46- 2 Planned EPU Power Ascension Testing

(*) Test Number	Title	Test Description
		used for temporary backup situations and low power. The feedwater control system in three element control mode should be adjusted, not only for stable operational transient level control (i.e., decay ratio), but also for stable steady-state level control (i.e., minimize reactor water limit cycles). In single element control mode, the system adjustments must achieve the operational transient level control criteria, but for steady state level control the temporary backup nature of this mode should be considered. For tests calling for manual flow step changes, at each test condition the feedwater control system is placed in a manual/auto configuration (i.e., one feedwater pump in manual and the others in automatic controlling water level). Preferably, the flow step changes are made by inserting the step demand change into the feedwater pump controller in manual or alternately by changing the setpoint of that controller in accordance with the following setpoint change sequence expressed in percent of feedwater pump flow. After completion of testing on one controller, the manual/auto configuration is switched and the sequence is repeated on the other controllers. <table><tr><th>Sequence</th><th>Flow Change</th></tr><tr><td>1</td><td>Step increase of 5%</td></tr><tr><td>2</td><td>Step decrease of 5%</td></tr><tr><td>3</td><td>Step increase of 10%</td></tr><tr><td>4</td><td>Step decrease of 10%</td></tr></table>The 5% flow step changes are informational and recommended to demonstrate the feedwater turbine response prior to performing the formal test flow step changes (i.e., 10%). The results from the smaller informational flow steps are utilized to anticipate the responses to the formal demonstration test, so that any effects on the reactor may be anticipated (i.e., level changes, power increases). <u>Feedwater Flow Calibration</u> Feedwater Flow and Steam Flow scaling and calibration for EPU conditions is verified by comparing indicated flow data to calculated feedwater flow and steam flow. <u>Maximum Feedwater Runout Capability</u> During the EPU power ascension, pressure, flow and controller data is gathered on the feedwater system

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Table 46- 2 Planned EPU Power Ascension Testing

(*) Test Number	Title	Test Description
		performance. This measured data is compared against expected values, which are based on information such as pump performance curves, turbine speeds, feedwater system flows and vessel dome pressure. This data is used to predict Feedwater flow run-out at EPU conditions and to verify that the run-out flow does not exceed that used in the transient analyses.
EPU-24	**TURBINE VALVE SURVEILLANCE**	This test will determine the maximum power at which surveillance testing of TSVs and Bypass Valves (BPVs) can be performed without a scram or isolation. Perform the surveillance test on at least one TSV and BPV in each EPU test condition. Additional valves have also been tested in some test conditions, as necessary, such that all valve surveillances are completed throughout the EPU test program. After each EPU test condition's test, margins to scram / trip setpoints for each valve type are determined and margins projected to the next test condition. Maintain a plot of power versus the peak variable values along the MELLLA Boundary to aid in projecting and evaluating the margins. The EPU surveillance test power level is the highest power level where all margins remain acceptable. Scram / trip margins may be more conservative values as appropriate for operational practices and preferences (e.g., APRM flow biased simulated thermal power rod block alarm not received during the tests). Each test is manually initiated, valve stroked, and reset in accordance with the current valve surveillance procedure. Note: TCVs are excluded from this test. Physical characteristics of the TCVs impair test testing above 95 % CLTP.

(*) The listed Test Number is for ease of referencing test descriptions between Tables 46-1 and 46-2. Actual test numbers will be assigned in accordance with TVA's administrative procedures.

Table 46- 2 Planned EPU Power Ascension Testing

(*) Test Number	Title	Test Description
EPU-100	**VIBRATION MONITORING**	During the EPU power ascension, designated piping points (i.e., location and direction) in selected systems (main steam, feedwater, etc.) will be monitored for vibration. Vibration monitoring points will be designated based on EPU piping vibration analysis and engineering judgment. Monitoring points may be coincidental with those in the initial startup piping vibration test or be selected as those points with the highest predicted vibration. Alternately, vibrations monitoring points can be coincidental with exposed piping attachments provided that acceptance criteria is established for those points based piping system vibration analysis. Vibration measurements taken above CLTP will permit a thorough assessment of the effect of the EPU in comparison to any previous piping vibration analysis or evaluation (See Attachment 45). The replacement steam dryer monitoring program is discussed in Attachment 40.
EPU-101	**EPU PLANT PARAMETER MONITORING AND EVALUATION**	Routine measurements of the power-dependent parameters from systems and components, affected by the EPU, are taken at < 95% and 100% of CLTP. Power increase is in incremental steps of 5% or less to ensure a careful, monitored approach to maximum EPU power. Power-dependent parameters are calculated using accepted methods to ensure current licensed and operational practices are maintained. Measured and calculated power-dependent parameters are utilized to project those values at the next power level step prior to increasing to the next test condition. Each step's projected values will be evaluated to have satisfactorily confirmed its actual values before advancing to the next step and the final increase to maximum EPU power.

(*) The listed Test Number is for ease of referencing test descriptions between Tables 46-1 and 46-2. Actual test numbers will be assigned in accordance with TVA’s administrative procedures.