NEXTera ENERGY
POINT BEACH

May 6, 2010

NRC 2010-0049
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

License Amendment Request 261
Extended Power Uprate
Response to Request for Additional Information

References:
(1) FPL Energy Point Beach, LLC letter to NRC, dated April 7, 2009, License Amendment Request 261, Extended Power Uprate (ML091250564)
(2) NRC letter to NextEra Energy Point Beach, LLC, dated April 1, 2010, Point Beach Nuclear Plant, Units 1 and 2–Request for Additional Information from Quality and Vendor Branch Re: Extended Power Uprate (ML100820563)

NextEra Energy Point Beach, LLC (NextEra) submitted License Amendment Request (LAR) 261 (Reference 1) to the NRC pursuant to 10 CFR 50.90. The proposed license amendment would increase each unit's licensed thermal power level from 1540 megawatts thermal (MWt) to 1800 MWt, and revise the Technical Specifications to support operation at the increased thermal power level.

Via Reference (2), the NRC staff determined that additional information was required to enable the staffs continued review of the request. Enclosure 1 provides the NextEra response to the NRC staffs request for additional information. Enclosure 2 provides LAR 261 Attachment 5, Appendix C, Matrices 12 and 13, which were inadvertently omitted from Reference (1). Enclosure 3 contains a revised Attachment 5, Section 2.12 of Reference (1).

This letter contains no new Regulatory Commitments and no revisions to existing Regulatory Commitments.

The information contained in this letter does not alter the no significant hazards consideration contained in Reference (1) and continues to satisfy the criteria of 10 CFR 51.22 for categorical exclusion from the requirements of an environmental assessment.

NextEra Energy Point Beach, LLC, 6610 Nuclear Road, Two Rivers, WI 54241

In accordance with 10 CFR 50.91, a copy of this letter is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on May 6, 2010.

Very truly yours,

NextEra Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosures

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

## ENCLOSURE 1

## NEXTERA ENERGY POINT BEACH, LLC
## POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

## LICENSE AMENDMENT REQUEST 261
## EXTENDED POWER UPRATE
## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

The NRC staff determined that additional information was required (Reference 1) to enable the Quality and Vendor Branch to complete the review of License Amendment Request (LAR) 261 (Reference 2). The following information is provided by NextEra Energy Point Beach, LLC (NextEra) in response to the NRC staff's request.

**EQVB 2.12-1**

*Section 1.0 of Attachment 5 states that the PBNP evaluations have been formatted and documented in accordance with the template and criteria provided in RS-001. However, the section titled "Scope and Associated Technical Review Guidance, Matrix 12," is not included in Appendix C of Attachment 5. Revise Appendix C of Attachment 5, as applicable, to include this reference.*

**NextEra Response**

Matrices 12 and 13 were inadvertently omitted from LAR 261, Attachment 5, Appendix C. Enclosure 2 provides Matrices 12 and 13 of Appendix C.

**EQVB 2.12-2**

*Section 2.12.1.2 of Attachment 5 states that the licensee has benefited from industry operating experience gained from discussions with other recently uprated PWRs (e.g., Ginna and Kewanee) and the Institute of Nuclear Power Operations. However, in Section 2.12.1.2.6, "Justification for Exception to Transient Testing," of Attachment 5, a discussion of such industry or PBNP plant-specific operating experience is not provided. Such information may be considered by the Nuclear Regulatory Commission (NRC) staff to support the basis for the licensee's request not to perform certain transient tests (e.g., Tests 14, 33 and 35) as part of the proposed extended power uprate (EPU) power ascension and testing plan (PATP). The licensee's primary basis for not performing such transient testing as part of the proposed EPU License Amendment Request (LAR) appears to rely solely on an analytical justification using LOFTRAN. Revise Attachment 5, as applicable, to include a discussion of such industry and plant-specific operating experience relative to the NRC staff's review criteria discussed in Section III.C.2 of Standard Review Plan 14.2.1.*

**NextEra Response**

This question asks for a discussion of the operating experience gained from recently uprated pressurized water reactors, in particular, those experience records used to justify not re-performing specific power-escalation tests, including Test 14, Steam Dump Control System; Test 33, Plant Trip; and Test 35, Load Swing and Load Reduction Test.

To support the analysis and conclusions stated in LAR 261 (Reference 2), Section 2.12.1.2.6, submittals and startup reports from other Westinghouse plant uprates were reviewed, with particular attention to similar 2-loop designs. The discussion of industry operating experience in power ascension testing results compared to analysis results using the LOFTRAN computer code is contained in Enclosure 3.

**EQVB 2.12-3**

*Section 2.12.1.1, of Attachment 5, states that detailed information regarding performance of the original PATP was provided to the Atomic Energy Commission in the 1971-1973 timeframe. However, information is not provided in the LAR regarding the details and outcomes of such testing, specifically the scope of transient testing. Also, final safety analysis report Section 13.4, "Initial Testing in the Operating Reactor," and Table 2.12-2 of Attachment 5, do not provide sufficient information regarding such testing. Revise Section 2.12.1.1, as applicable, to include a discussion regarding performance of such original PATP tests, with emphasis on the performance of transient Tests 14, 33 and 35. Such information is necessary for the NRC staff to consider in support of the licensee's justification for not performing certain startup tests, including transient tests, as part of the licensee's proposed EPU PATP.*

**NextEra Response**

This question asks for a description of the testing report from the original plant startup, specifically the power-escalation Test 14, Steam Dump Control System; Test 33, Plant Trip; and Test 35, Load Swing and Load Reduction Test. The successful results of this testing are presented in support of a justification for not performing transient tests following the PBNP uprate. The discussion of PBNP-specific operating experience in power ascension testing during original plant startup is contained in Enclosure 3.

**References**


(1) NRC letter to NextEra Energy Point Beach, LLC, dated April 1, 2010, Point Beach Nuclear Plant, Units 1 and 2 – Request for Additional Information from Quality and Vendor Branch Re: Extended Power Uprate (ML100820563)

(2) FPL Energy Point Beach, LLC letter to NRC, dated April 7, 2009, License Amendment Request 261, Extended Power Uprate (ML091250564)

# ENCLOSURE 2

## NEXTERA ENERGY POINT BEACH, LLC
## POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

## LICENSE AMENDMENT REQUEST 261
## EXTENDED POWER UPRATE
## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

## SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE
## APPENDIX C
## MATRICES 12 AND 13

2 pages follow

# APPENDIX C
# MATRIX 12

## SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

### Power Ascension and Testing Plan

| **Area of Review (NRC Review Criteria)** | **Acceptance Criteria (PBNP Specific GDCs)** | **Other Guidance** |
|---|---|---|
| Approach to EPU Power Level and Test Plan<br>LR Section 2.12.1 | | |
| 10 CFR 50, Appendix B, Criterion XI | Regulatory Guide 1.68 | FSAR Chapter 13, Table 13.2-1 |

**APPENDIX C**
**MATRIX 13**

## SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

### Risk Evaluation

| Area of Review (NRC Review Criteria) | Acceptance Criteria (PBNP Specific GDCs) | Other Guidance |
|---|---|---|
| Risk Evaluation of EPU LR Section 2.13.1 | | |
| Regulatory Guide 1.174<br>RIS 2001-02 | Generic Letter 88-20 | |

# ENCLOSURE 3

## NEXTERA ENERGY POINT BEACH, LLC
## POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

## LICENSE AMENDMENT REQUEST 261
## EXTENDED POWER UPRATE
## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

## LICENSE AMENDMENT REQUEST 261
## REVISION TO ATTACHMENT 5, SECTION 2.12

32 pages follow

## 2.12 Power Ascension and Testing Plan

### 2.12.1 Approach to EPU Power Level and Test Plan

#### 2.12.1.1 Regulatory Evaluation

The purpose of the EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions. The review included an evaluation of:

- plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance,
- transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and
- the test program's conformance with applicable regulations.

The NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service. Specific review criteria are contained in SRP Section 14.2.1.

**Current PBNP Licensing Basis**

The initial startup test program at the PBNP is described in FSAR Chapter 13, Objectives and Scope. FSAR Table 13.2-1, Preoperational Tests, lists the initial plant startup tests performed to place equipment in service.

After the operating characteristics of the reactor and plant had been verified by initial verification and low power tests, a program of power escalation in successive stages was undertaken to bring the plant to its full rated power level. Both reactor and plant operational characteristics were closely examined at each stage and the relevance of the safeguards analysis was verified before escalation to the next programmed level was effected. Based upon data obtained from low power tests, the first escalation was to approximately 40% reactor thermal power. The data at each level was analyzed to determine what indications would be when reactor thermal power was at the next escalation level. Succeeding levels were at approximately 70% and 100% core thermal power.

Reactor physics measurements were made to determine the magnitudes of the power coefficient of reactivity, of xenon reactivity effects, of Rod Cluster Control (RCC) control group differential worth and of relative power distribution in the core as functions of power level and RCC control group position.

Concurrent determinations of primary and secondary heat balances were made to ensure that the several indications of plant power level were consistent and to provide bases for calibration of the power range nuclear channels. The ability of the reactor control and protection system to respond effectively to signals from plant primary and secondary instrumentation under a variety of conditions encountered in normal operations was verified.

At prescribed power levels the response characteristics of the reactor coolant and steam systems to dynamic stimuli were evaluated. The responses of system components were measured for 10% loss of load and recovery, 50% loss of load and recovery, turbine trip, loss of flow and trip of a single RCC unit.

A series of load follow tests were performed at selected power level escalation steps and after rated power level had been achieved. The results of these tests gave actual reactor and plant behavior under operating conditions and were used to verify predicted load follow capabilities.

Adequacy of radiation shielding was verified by gamma and neutron radiation surveys inside the containment and throughout plant buildings and yard areas.

The sequence of tests, measurements and intervening operations were prescribed in the power escalation procedures together with specific details relating to the conduct of the several tests and measurements. The measurements and test operations during power escalation were similar to normal plant operations.

Detailed information on the above power ascension testing is provided in the summaries submitted to AEC, March 1971 for Unit 1, and the two submittals for Unit 2, September 1972 and October 1973 covering the testing at the 20 and 100% power levels, respectively. Because testing of the two units was similar, this report primarily refers to Unit 1 power ascension and testing.

Transient testing, including trips from various power levels, ramp load changes at 5%/minute, and 10% step load changes were performed during the initial startup of both PBNP units. Specific testing on Unit 2 included the following:

- A 10% step load change at low power
- A 10% step load change from 90% to 80% and from 80% to 90%
- A ramp load change at 5%/minute from 92% to 41% and back to 92%
- Two turbine loss of load/turbine overspeed tests from 30% power
- One turbine loss of load/turbine overspeed test from 70% power

Transient testing performed during the Unit 1 startup was similar to the above, and included six overspeed (loss of load) tests initiated from 40%, 55%, 70% (3 tests), and 90% power.

For all of these tests, plant response was consistent with the setpoint studies. Following unit trips on both units, controls for feedwater, pressurizer pressure and level, and steam pressure responded as well or faster than predicted. During step load increases, feedwater flow was initially reduced due to swell, but then increased as expected. A turbine overspeed trip test from 40% power on Unit 1 was successful in not resulting in a reactor trip. Following a trip from 70% power on Unit 1, $T_{avg}$ decreased in about one-half the predicted time, attributable to the effectiveness of the steam dumps.

Performance of plant controls demonstrated during plant startup and enhanced by upgrades in control systems installed and proven over the years of plant operation are expected to continue to operate reliably at EPU conditions. Additional control modifications to be in place and tested before uprate implementation (e.g., new digital feedwater heater level controls and new digitally-controlled operators on the feedwater regulating valves) can be reasonably expected to mitigate transients at least or more effectively as those in the original plant. Repeating such original plant startup testing would place unnecessary stress and cycling on unit equipment. Therefore, performing these tests would not confirm any new or significant aspect of performance at EPU conditions not already demonstrated through analysis, operating experience, or routine plant operations and the risk of performing such tests should not be incurred.

The current licensed reactor power level for PBNP is 1540 MWt. The 1.4% Measurement Uncertainty Recapture (MUR) increase from the original licensed power level was approved by NRC in the Safety Evaluation dated November 29, 2002. Testing at the 1540 MWt reactor power level was completed; the approach to the 1540 MWt power level was undertaken carefully, with calorimetric measurements used to install the revised $\Delta T$ and nuclear instrumentation reactor protection setpoints. Plant operating conditions were verified acceptable and in accordance with predicted analyses and design documentation.

### 2.12.1.2Technical Evaluation

#### 2.12.1.2.1 Introduction

PBNP is currently proposing an Extended Power Uprate (EPU) to increase core thermal power to 1800 MWt. This uprate involves changes to the plant configuration to accommodate the higher reactor power limit as well as the larger steam and feedwater flows commensurate with the power increase. As a result of these changes, testing is required to ensure that the plant can be operated safely in its uprated condition.

#### 2.12.1.2.2 Background

The proposed EPU at PBNP will result in the reactor operating at a new core thermal power of 1800 MWt. The current licensed core thermal power is 1540 MWt. PBNP has significant operating experience at its current operating condition. PBNP is a Westinghouse two-loop design, and power levels close to the proposed EPU level have been successfully achieved by similar Westinghouse two-loop design plants, such as Kewaunee and Ginna, with no adverse affects.

In a PWR, the largest change in system operating parameters occurs in the secondary side where mass flow is increased commensurate with the uprate. Minor changes also occur in primary side temperatures to provide additional heat transfer in the steam generators. At PBNP, the main steam and condensate/feedwater flows will increase by more than 20%. The full power main steam operating pressure will be slightly less than for current operation, however, reactor coolant operating average temperature, $T_{avg}$ will be increased to 576°F.

In order to accommodate this new thermal power, changes in plant operating parameters have to occur. However, it has been found that the fundamental operating characteristics of an uprated plant remain consistent with the operating characteristics prior to the uprate, and also consistent with other similar units that have been uprated. This means that pre-uprate plant operating experience and industry operating experience provide insight to the viability of a plant uprate.

This operating experience will be incorporated into the detailed test plan and controlling procedures.

Several plant modifications are required to support power operation at the proposed uprated core thermal power. Post-modification testing of these modifications will be performed to ensure proper installation. Additionally, system surveillance tests will be performed as required to verify that the modifications meet applicable performance criteria. Integrated plant analyses were performed to define the performance criteria of the various plant modifications necessary to accommodate the uprated power. The results of these analyses, coupled with the evaluation of plant data acquired during power ascension, are used, in part, in lieu of large transient testing to verify that the plant systems are capable of performing safely in the uprated condition.

The EPU testing program will also draw on the results of the original startup and test program and applicable industry experience as a means of ensuring safe operation at the new core thermal power level. Comparisons will be made between recent operating data and the data that will be gathered during the uprate testing to ensure that the results are reasonable. Additionally, PBNP has years of operating experience at the current licensed power level such that system interactions are well known. Ginna and Kewaunee have uprated to a core thermal power levels that are similar to the PBNP EPU power level (1800 MWt) and have operated successfully at the new power level. PBNP has established communication with Ginna and Kewaunee in order to benefit from their power uprate experience.

In addition to Kewaunee and Ginna, PBNP has benefited from industry operating experience in power uprate implementation from several industry sources, including INPO. The PBNP test plan is based on industry operating experience pertaining to power uprate and has used this experience in the formulation of expected system interactions, design of EPU modifications, determination of control system settings and setpoints, and development of post-modification and power ascension test plans. For example, PBNP has learned lessons from the industry regarding vibration and vibration monitoring, iso-phase bus duct cooling and air flow, turbine controls, feed/condensate/drain system flows and pressure drops, feedwater heater performance and reactor control system setpoints.

In summary, the proposed EPU testing program is comprised of a mixture of power ascension monitoring, post-modification testing and analytical evaluation and transient testing, to ensure that the plant can operate safely at its new uprated core thermal power. The following sections describe the proposed PBNP Power Ascension Testing Program and demonstrate that the proposed testing program contains all of the necessary elements to assure safe operation at the uprated power level.

### 2.12.1.2.3 Proposed Power Ascension Test Plan

#### 2.12.1.2.3.1 General Discussion

The development of the power uprate test program is based on a review of similar test programs performed at other plants and the outputs of various system and integrated plant analyses performed in support of the EPU. Additionally, FSAR Chapter 13, Section 13.4, Initial Testing in the Operating Reactor, describes the test methodology used during the original power ascension was also reviewed. This review was augmented by a review of the actual original power ascension test summaries, in addition to the MUR test documentation.

Prior to the commencement of power ascension testing, the EPU Test Program will require the completion of numerous activities, which include:

- Review and revision of applicable plant operating procedures, administrative procedures, and surveillance test procedures, calibration procedures, chemical and radiological procedures, and other similar procedures.
- Review and revision of computer software programs as required to support the power uprate test program and the new EPU power level.
- Incorporation of applicable plant instrumentation setpoint changes and recalibration of instrumentation as required.
- Implementation and successful post-modification testing of all required plant modifications.
- Review of Temporary Modification logs and Operable but Degraded or Nonconforming conditions to assure there is no impact on the ability of the affected equipment to support uprate, and that uprate will not have an adverse impact on an existing plant condition.

Additionally, commitments which are the result of the EPU License Amendment Request, the NRC EPU Safety Evaluation (SE), and other actions associated with the PBNP EPU implementation, will be verified as being complete, included in the Power Ascension Testing Program, or evaluated as not impacting power ascension.

The EPU Power Ascension Test Program will be developed to verify the following:

- Plant systems and equipment affected by EPU are operating within design limits
- Nuclear fuel thermal limits are maintained within expected margins and the core is operating as designed
- Steam generator water level control is stable with adequate control margin to allow for anticipated transients
- Reactor control systems are stable and capable of maintaining reactor parameters within acceptable limits
- Moisture Separator Reheater (MSR) and feedwater heater drains and level control are stable
- System radiation levels are acceptable and stable
- General area and local environmental conditions are acceptable

The EPU test program consists of a combination of normal startup and surveillance testing, post-modification testing, and power ascension testing deemed necessary to support acceptance of the proposed EPU. The following system and equipment testing has been evaluated for inclusion into the EPU test plan and test program:

- Initial startup testing identified in FSAR Table 13.2 -1, Preoperational Testing (See Table 2.12-1, PBNP Extended Power Uprate Power Ascension Test Plan, and Table 2.12-2, EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests, for EPU planned testing)

- Pre-modification baseline testing
  - Turbine performance test (high-pressure turbine replacement)
  - Piping vibration monitoring (balance of plant)
  - Monitoring of plant parameters
- Post-modification testing (as required and controlled by the design change process). See LR Section 1.0, Introduction to the Point Beach Nuclear Plant Units 1 and 2 Extended Power Uprate Licensing Report for list of Plant Modifications
- Power ascension testing
  - Monitoring of plant parameters
  - Piping vibration monitoring (balance of plant)

Plant modifications will be implemented at PBNP in order to achieve and support the EPU rated power: they are controlled by administrative procedures which provide configuration control, installation instructions, and testing requirements. Post modification testing verifies satisfactory performance of the modification in accordance with the design documentation. The performance of post-modification testing is addressed by existing programmatic controls within the design modification process. Functional and operational post modification testing will be performed for each modification to verify satisfactory installation and performance.

2.12.1.2.3.2 EPU Power Ascension Test Plan and Test Plateaus

Performance in accordance with expectations based upon analyses and operating experience of similar equipment will be established. Acceptance criteria will be established for each plant parameter determined to be included in the "monitored parameter list." See Table 2.12-1, PBNP Extended Power Uprate Power Ascension Test Plan, for an overview of the planned power escalation testing. Industry operating experience as well as consultation with PBNP engineering personnel and industry experts at vendors with significant power uprate testing experience will be used in the selection process.

During the EPU startup, power will be increased in a slow and deliberate manner, stopping at pre-determined power levels for steady-state data gathering and formal parameter evaluation. These pre-determined power levels are referred to as Test Plateaus. The typical post-refueling power plateaus will be used until the current (1540 MWt) full power condition is attained at approximately 85% of the EPU power level (1800 MWt). Above this power level, smaller intervals between test plateaus will be established, with a concurrent higher frequency of data acquisition. The summary of the Power Ascension Test Plan is provided in Table 2.12-1, PBNP Extended Power Uprate Power Ascension Test Plan.

Prior to exceeding the current licensed core thermal power of 1540 MWt, the steady-state data gathered at the pre-determined power plateaus, and transient data gathered during the specified transient tests at lower power, as well as observations of the slow, but dynamic power increases between the power plateaus, will allow verification of the performance of the EPU modifications. In particular, by comparison of the plant data with pre-determined acceptance criteria, the test plan will provide assurance that unintended interactions between the various modifications have not occurred such that integrated plant performance is adversely affected.

Once at approximately 85% of EPU power, (1540 MWt), power will be slowly and deliberately increased through 5 additional Test Plateaus, each differing by approximately 3% of the EPU rated thermal power. Both dynamic performance during the ascension and steady-state performance for each Test Plateau will be monitored, documented and evaluated against pre-determined acceptance criteria.

Following each increase in power level, test data will be evaluated against its performance acceptance criteria (i.e., design predictions or limits). If the test data satisfies the acceptance criteria then system and component performance will be considered to have complied with their design requirements.

In addition to the steady-state parameter data gathered and evaluated at each Test Plateau, the dynamic parameter response data gathered during the ascension between test plateaus will also be thoroughly reviewed. Of major concern is the overall stability of the plant, and potential changes in transient responses that may arise due to the EPU modifications to the secondary systems.

Hydraulic interactions between the new condensate and new feedwater pumps, and modified feed regulating valves, as well as the impact of the higher main feed flow and the associated increased piping pressure loss will be evaluated. Individual control systems such as steam generator level control and moisture separator and feedwater heater drain level control will be optimized for the new conditions as required. It is anticipated that the proposed tests will adequately identify unanticipated adverse system interactions and allow them to be corrected in a timely fashion prior to full power operation at the uprated conditions.

Table 2.12-1, PBNP Extended Power Uprate Power Ascension Test Plan, provides a summary of the Power Ascension Test Plan. Table 2.12-2, EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests, provides a summary of the original startup testing, and a brief comparison with the proposed power ascension test plan. Further, Table 2.12-2, EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests, provides justification for not repeating several of the original tests during the proposed EPU test plan.

#### 2.12.1.2.3.3 Acceptance Criteria

The acceptance criteria for the PBNP power ascension test program will be established as discussed in Regulatory Guide 1.68.

Level 1 acceptance criteria are values for process parameters assigned in the design of the plant that are safety significant. If a Level 1 criterion is not satisfied, the power ascension will be stopped and the plant will be placed in a condition that is safe based upon prior testing. The power escalation test procedure and Technical Specifications will provide direction for actions to be taken to assure the plant is safe and stable. Resolution of the issue that resulted in not meeting the Level 1 criterion must be resolved by equipment changes or through engineering evaluation, as appropriate. Following resolution, the applicable test portion must be repeated to verify that the Level 1 requirement is satisfied. A description of the problem must be included in the report documenting successful completion of the test.

Level 2 acceptance criteria are values that relate to plant functions or parameters that are not safety significant. If Level 2 criteria are not met, the Power Ascension Test Plan may continue. Investigation of the issue that resulted in not meeting the Level 2 criterion may continue in parallel with the power escalation. These investigations would be handled by existing plant processes and procedures.

For the PBNP Power Ascension Test Plan specific Level 1 and 2 acceptance criteria will be established and incorporated into the Power Ascension Test Procedure, (See Attachment 4, Item 24).

2.12.1.2.3.4 Vibration Monitoring

A Piping and Equipment Vibration Monitoring Program, including plant walkdowns and monitoring of plant equipment, will be established to ensure that steady state flow induced piping vibrations following EPU implementation are not detrimental to the plant, piping, pipe supports or connected equipment.

Observed piping vibrations will be evaluated to ensure that damage will not result. The predominant way of assessing these vibrations is to monitor the piping during the plant heat up and power ascension. The methodology to be used for monitoring and evaluating this vibration will be in accordance with ASME OM-S/G-2003.

The scope of the Piping and Equipment Vibration Monitoring Program includes any accessible lines that will experience an increase in their process flow rates. Any branch lines attached to these lines (experiencing increased process flows) will also be monitored as experience has shown that branch lines are susceptible to vibration-induced damage. The scope of the Piping and Equipment Vibration Monitoring Program includes the following systems:

- Main, and Reheat Steam (outside of containment)
- Steam Generator Blowdown
- Feedwater System (outside of containment)
- Condensate System
- Feedwater Heater Vents Relief and Miscellaneous Drains
- Feedwater Heater Drains
- Extraction Steam [and TG Gland Seal and Exhaust]
- Turbine Plant Miscellaneous Drains

The main steam and feedwater piping inside containment is not readily accessible for performing vibration monitoring during power ascension. This piping inside containment is not considered to be a target area for the following reasons:

- The main steam and feedwater piping is well supported and seismically designed.

- The piping is large diameter, not overly flexible, with large diameter bends and few elbows.
- There are no long cantilever branch lines or branch lines with heavy unsupported valves.
- There is no history of vibration problems in these lines at PBNP.
- Operating experience from other 2-loop Westinghouse-designed stations for EPU licensed power levels and which have similar piping and support designs has not identified a history of vibration problems with these lines.
- Review of operating experience at recent EPU stations has not identified significant vibration in these systems inside containment which would have been a safety or failure concern.

Reactor Coolant System piping (RCS) is not included in the scope of this vibration monitoring program as the system does not experience a significant change in flow due to uprate even though there may be minor RCS mass and volumetric flow changes depending on location due to density distribution changes.

The following equipment monitoring will be included:

- Feedwater and Condensate Pumps
- Feedwater and Condensate Motors
- Heater Drain Pumps
- Main Turbine Generator

The program scope will also include any lines or equipment within the monitored systems that have been modified or otherwise identified through the PBNP action report system as having already experienced vibration issues.

The piping and equipment within the scope of the vibration monitoring program will be observed at several different plant operating conditions. The first observations will be conducted prior to the shutdown in which the EPU will be implemented. Data from these observations will be used to develop a list of priorities for observation during the subsequent power escalation.

Subsequent observations will take place at each EPU Test Plateau, as described in Section 2.12.1.2.3.1 above. By comparing the observed pipe vibrations/displacements at various power levels with previously established acceptance Criteria, potentially adverse pipe vibrations will be identified, evaluated and resolved prior to failure.

#### 2.12.1.2.4 Comparison of Proposed EPU Test Programs to the Initial Plant Test Program

The following table (Table 2.12-2, EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests) provides a comparison of the original plant startup testing, as listed in FSAR Section 13.4, Initial Testing in the Operating Reactor, to the proposed Power Ascension Test Program. The table lists all tests performed during original power ascension regardless of power level at which they were performed. Included in the table are descriptions of the original test, listings of the original power level at which the test was performed, whether the test will be replicated as part of the Power Ascension Test Program, and the justification for why it is not performed (if it is not performed). Note that Table 2.12-1, PBNP Extended Power Uprate Power Ascension Test Plan provides more detail on specific data acquisition test plateaus.

### 2.12.1.2.5 Transient Analytical Methodology

Initiating Events are defined in ANSI N18.2 -1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. The conditions are:

- Condition I - Normal operation
- Condition II - Incidents of moderate frequency
- Condition III - Infrequent incidents
- Condition IV - Limiting faults

**Condition I Initiating Events**

Analyses and evaluations have been performed for the Condition I operating transients to assess the aggregate impact of the equipment modifications and setpoint changes for EPU conditions. These analyses and evaluations used the same principal computer code (i.e., LOFTRAN) that has been used in control system analyses for PBNP at current power conditions. The LOFTRAN computer code is described in WCAP-7907 P-A (LOFTRAN Code Description, April 1984) (Reference 1). The code has been approved by the NRC and has been used for many years for accident evaluations for Safety Analysis Reports, and for control system performance and equipment sizing studies.

LOFTRAN has been used in the analysis of Condition I initiating events on PBNP as well as on other Westinghouse designed nuclear power plants. The NRC Safety Evaluation (SE) included in WCAP-7907-P-A describes the LOFTRAN verification process performed by Westinghouse for transients including reactor trip from 100% power, 100% load reduction, and step load changes. The verification process consisted of comparison of LOFTRAN results to actual plant data and to other similar thermal-hydraulic programs. The LOFTRAN verification process also included analysis of a R. E. Ginna steam generator tube rupture (SGTR) event, where comparison of the LOFTRAN results to available plant data demonstrated the ability of LOFTRAN to analyze the SGTR event.

The NRC SER included in WCAP-7907-P-A concludes that the data comparisons and the results comparisons provided by Westinghouse demonstrate the ability of LOFTRAN to analyze the types of events for which it has been used in licensing safety analysis. In conjunction with its extensive use for many years, it has been used in evaluation of Condition I operating transients at many Westinghouse designed nuclear power plants including other similar Westinghouse designed 2-loop nuclear power plants currently operating at approximately 1775 MWt NSSS power.

The LOFTRAN computer code was used to analyze the following Condition I initiating events and Condition II turbine trip transient at PBNP at EPU conditions:

- Step load increase of 10% of full power from 90% to 100% power
- Step load decrease of 10% of full power from 100% to 90% power
- Large load reduction of 50% of full power from 100% power
- Turbine trip without reactor trip initiated from P-9 setpoint, (of uprated full power)

- Turbine trip from 100% power

Based on these limiting analyses run with LOFTRAN, the ramp load increase and decrease of 5% of full power per minute between 15% to 100% power was evaluated as being acceptable at the EPU conditions.

The LOFTRAN analysis inputs and models were updated as appropriate to incorporate the applicable EPU equipment modifications and setpoint changes as well as the EPU operating conditions. The analyses results showed that the plant responses to Condition I initiating events satisfied acceptance criteria and that the NSSS control system responses were stable. Furthermore, plant responses to Condition I initiating events were shown to have acceptable margins to reactor trip and engineered safety features actuation. The results of the analyses performed for Condition I initiating events at EPU conditions are reported in LR Section 2.4.2, Plant Operability. The plant responses to Condition I initiating events at EPU conditions are consistent with their characteristic responses based on operational and analytical experience on PBNP at the current power conditions as well as operational and analytical experience on other similar Westinghouse designed 2-loop nuclear power plants (Ginna and Kewaunee) currently operating at approximately the same NSSS power.

**Condition II, III, and IV Initiating Events**

Analyses and evaluations have been performed for the Condition II, III, and IV operating transients to assess the aggregate impact of the equipment modifications and setpoint changes for EPU conditions. Analysis inputs and models were updated as appropriate to incorporate the EPU equipment modifications and setpoint changes as well as the EPU operating conditions. These analyses results showed that the plant responses to Condition II, III, and IV initiating events satisfied acceptance criteria. The results of the analyses performed for Condition II, III, and IV initiating events at EPU conditions are reported in LR Section 2.8.5, Accident and Transient Analyses.

The dynamic plant responses to Condition II, III, and IV initiating events at EPU conditions with the EPU equipment modifications and setpoint changes are consistent with their characteristic responses based on operational and analytical experience at other similar Westinghouse designed 2-loop nuclear power plants (Kewaunee and Ginna) currently operating at approximately the same core thermal power.

**Natural Circulation**

Natural circulation capability for the PBNP EPU is evaluated using the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG) methodology. This method is used to estimate flow rates and core delta temperatures using core hydraulic resistance coefficients.

2.12.1.2.6 Justification for Exception to Transient Testing

PBNP has reviewed the recommendations of draft Standard Review Plan (SRP) for the EPU testing programs. As a result of this review, and a review of the original PBNP startup test program and recommendations from the NSSS and BOP vendors, PBNP concludes that no large load transient tests need to be performed as part of the EPU test program. This section discusses the justification for not performing the large transient tests.

**Justification for Exception - General**

PBNP is being modified to allow for operation at the process conditions associated with 1800 MWt core power level. The LOFTRAN computer code was used to evaluate plant response to Condition I and II initiating events at EPU conditions. The LOFTRAN computer code has been verified with respect to plant data and has been approved by the NRC for use in licensee safety analysis. The LOFTRAN verification process included comparison with plant data for transients including reactor trip from 100% power, 100% load reduction, and step load changes. The LOFTRAN verification process also included comparison with plant data for a steam generator tube rupture (SGTR) event that occurred at Ginna, where the comparison of the LOFTRAN results to available plant data demonstrated the ability of LOFTRAN to analyze the SGTR event. The code has been used by Westinghouse for accident evaluations for Safety Analysis Reports and for control system performance and equipment sizing studies. The application of the LOFTRAN computer code to PBNP considers any limitations included in NRC approval of the code along with plant-specific operating parameters and system configurations.

The LOFTRAN computer code has been used for PBNP for many years at the original and current power levels. In addition to its use on PBNP, it has also been used in evaluation of Condition I and II operating transients at many Westinghouse designed nuclear power plants including other similar Westinghouse designed 2-loop nuclear power plants. This use of LOFTRAN for analysis in a wide variety of different Westinghouse plants for various types of transients - both licensing/design basis analyses and for plant problem troubleshooting - has shown that this computer code can acceptably be used to predict the plant response, thereby negating the need to perform plant transient testing to validate the predicted code responses to large plant transients.

The LOFTRAN analysis inputs and models were updated as appropriate to incorporate EPU-related changes to parameter and setpoint values. Bounding inputs for design parameters were used as described in LR Section 1.1, Nuclear Steam Supply System Parameters. Analyses and evaluations were then performed for the NSSS control systems at EPU conditions. The NSSS control systems include the reactor (rod) control system, reactor coolant temperature ($T_{avg}$) control system, pressurizer level control system, pressurizer pressure control system, steam generator level control system, and steam dump control system. NSSS control systems setpoints are being revised as required to support EPU operations. Control systems including the rod control and $T_{avg}$ control system, pressurizer and level control system will have setpoints changed as described in LR Section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems.

NSSS control systems analyses were performed at EPU conditions for the following design basis Condition I operating transients and the Condition II turbine trip transient to demonstrate acceptable stability and setpoints:

1. 10% step load increase from 90% to 100% of uprated full power

2. 10% step load decrease from 100% to 90% of uprate full power

3. 50% load reduction from 100% of uprated full power

4. Turbine trip without reactor trip from P-9 setpoint (of uprated full power)

5. Turbine trip followed by a reactor trip from 100% of uprated full power

The NSSS control systems analyses assessed the aggregate impact of the applicable equipment modifications and setpoint changes at EPU conditions. The analyses results demonstrate that plant response to operations transients is acceptable, NSSS control systems responses are stable, and margins to reactor trip and engineered safety feature actuations are acceptable. Specifically, the performance of the rod control system and the steam dump control system is acceptable during both steady-state and transient operating conditions. The results also show that sufficient operating margins exist to reactor trip and engineered safety feature (ESF) actuation set points at EPU conditions with the NSSS control systems in the automatic mode. The NSSS control systems' pressure control components (i.e., pressurizer power operated relief valves, pressurizer spray valves, pressurizer heaters, and condenser steam dump valves) satisfy sizing requirements at EPU conditions and are acceptable for the analyzed transients.

These results are consistent with experience on several similar Westinghouse-designed, 2-loop nuclear power plants that use the LOFTRAN computer code for analysis of Condition I and II initiating events and operate at approximately the same NSSS power level as for PBNP at EPU conditions.

Of considerable benefit was the power escalation report from the R.E. Ginna Nuclear Power Plant (Ginna) (ML070380098), where several transient tests were performed, and shown to confirm LOFTRAN predictions and setpoint studies. At Ginna, power was uprated to 1775 MWt, which was essentially equivalent to Point Beach Nuclear Plant (PBNP). Transient tests were performed at an initial power level of 30% and at full power as described below. The results from these tests were used to support the conclusions stated below, that LOFTRAN analyses performed, resulting predictions for Ginna and similar predictions for PBNP are confirmed by the Ginna test results.

Load Swing Tests at Ginna

From an initial power level of 30%, a 10% load decrease at 1%/minute was initiated. After the unit became stable and data recorded, a 10% load increase at the same rate was initiated. The following was noted from the successful completion of this test:

- No reactor trip
- No power-operated relief valve (PORV) actuation
- No main steam safety valve opening
- Average reactor coolant temperature ($T_{avg}$), pressurizer level, pressurizer pressure and steam generator levels all remained within established acceptance criteria for the transient

Only steam generator pressure fell below the acceptance value, which was explained to be acceptable because pressure was at the low end of the band at the beginning of the test, and steam pressure is often low during startup from refueling. This condition had no effect on test results.

After reaching 100% power, two ramp load change tests were performed. A 3% down and up, followed by a 10% down and up; both performed using a similar procedure with similar acceptance criteria to that used for the 30% test. The acceptance criteria were met satisfactorily in both tests. Initial steady-state and response data were recorded that included reactor power, $T_{avg}$, pressurizer pressure and level, and steam generator pressure and level.

The load swing tests described here verified that pressurizer pressure and level control, rod control, $T_{avg}$/reference average reactor coolant temperature ($T_{ref}$) following, steam generator level control, feedwater flow, condensate system pressure and turbine controls all functioned properly and consistent with the analyses.

Turbine Trip Test at Ginna

A manual turbine trip from 30% was performed. The test exercised control systems including rod control, steam dump control, pressurizer level and pressure control, and steam generator water level control. The test was preceded by preparations that included calibration checks and post-modification testing of control systems. The manner in which the control systems responded to the power and temperature mismatch as a result of the turbine trip were verified, including the ability of the control systems to achieve stable plant conditions in an acceptable range. Satisfied acceptance criteria included a demonstration that the plant dynamic response was stable and converged on a range that supports safe operation at low power and the following specific criteria:

- No reactor trip
- No PORV actuation
- No main steam safety valve opening
- Turbine stop valves closed
- Steam dump valves operated to control steam pressure
- $T_{avg}$ remained in acceptable band and stabilized at 550°F
- Pressurizer pressure and pressurizer level remained in acceptable band
- Main feedwater regulating valves restored steam generator levels to programmed range
- Reactor power decreased to 14%; operators were able to place rod control in manual and control power between 10 - 15%

Immediately following a turbine trip, steam dump controls were confirmed to be operating. $T_{avg}$ was confirmed to remain within the acceptable range of 545 to 579°F and stabilized at approximately 550°F.

At Ginna, setpoint changes were implemented in steam dump controls that were verified by the turbine trip test. Steam dump setpoints at PBNP are not being changed. Therefore, consistency of the data to the analyses and setpoint studies confirms the adequacy of the steam dump controls for PBNP. In a similar manner to Ginna, any control setpoint changes are validated through calibration prior to plant startup.

Satisfactory completion of the turbine trip test fulfilled the purpose of the pressurizer level control test, pressurizer pressure control test, and steam dump test performed during original plant startup testing for Ginna. Test documentation provided data to be used when appropriate to tune both the simulator and engineering design models. The test also provided an opportunity to gain operator experience with a load rejection transient under controlled conditions that may then be used to adjust operating procedures when necessary. By ensuring confidence in the parameters predicted by the analyses and programmed into the PBNP training simulator, operators benefit from valid simulator training exercises.

The Ginna operating experience described above in conjunction with the PBNP-specific operating experience described in Section 2.12.1.1, and the results of LOFTRAN predictions of plant response at EPU conditions support the request not to perform certain transient tests, including tests of the steam dump control system, plant trips, and load swings and reductions. As stated in LAR 261, Section 2.12.1.2.6, no new thermal-hydraulic phenomena are introduced by either the physical modifications or the changes in operating conditions and that no new system dependencies or interactions are being introduced by the changes. Therefore, performing these tests would not confirm any new or significant aspect of performance at EPU conditions not already demonstrated through analysis, operating experience, or routine plant operations and the risk of performing such tests should not be incurred.

Other process parameter changes being made to accommodate the power increase are within the design capability of the related systems, or necessary upgrades are being installed. Therefore, no new thermal-hydraulic phenomena are introduced by either the physical modifications or the changes in operating conditions. Furthermore, the results of the analyses indicate that no new system dependencies or interactions are being introduced by the changes.

As discussed above, the aggregate impact of the EPU equipment modifications and setpoint changes on the dynamic plant response of PBNP for Condition I and II initiating events at EPU conditions was assessed through analyses and evaluations. These analyses and evaluations used the LOFTRAN computer code, which has been verified and approved by the NRC. The extent of the aggregate impact of the EPU equipment modifications and setpoint changes on dynamic plant response is such that it can be adequately addressed through analyses and evaluations. It is accepted practice to use analyses and evaluations to assess the aggregate impact of these types of equipment modifications and setpoint changes on PBNP as well as on other Westinghouse designed nuclear power plants.

Therefore, performing the load transient tests identified above would not confirm any new or significant aspect of performance not already demonstrated through analysis, by previous operating experience or routinely through plant operations. The following provides a description of the load transient tests and justification for exception.

Justification for Exception - Specific

Electrical Load Loss and Load Swings

The net electrical load loss from below Permissive P-9 Setpoint and the load reduction of 50% load at high power are tests to demonstrate that the control systems act together to prevent a reactor trip and also prevent the opening of the main steam safety valves (MSSVs). In particular, the test demonstrates that the rod control, steam dump and pressurizer pressure and level control systems act together to control the NSSS response to within design limits and the reactor trip setpoints. An analysis of a 50% load reduction from 100% EPU power was performed using the LOFTRAN code as described in LR Section 2.4.2, Plant Operability. This analysis demonstrates that the PBNP response to a 50% load reduction will not cause a reactor trip and will not cause MSSVs to open. An analysis of a loss of load from the Permissive P-9 setpoint was also performed at EPU conditions to demonstrate that the PBNP response to step load decrease from below the P-9 setpoint will not cause a reactor trip and will not cause the pressurizer power operated relief valves (PORVs) to open.

There are no major hardware modifications planned for NSSS components as part of the EPU that would affect the plant transient response. Since the NSSS control system functional design and hardware are not impacted and the analyzed 50% load reduction Condition I operating transients show acceptable stability, setpoints, and margin to reactor trip and ESF actuation, the NSSS control systems are acceptable for operation at full power EPU conditions. Analysis of the 50% load reduction provides a bounding justification for not performing 10% load swings either as step or ramp changes. A reactor trip, or the potential for a reactor trip, from high power level results in an unnecessary plant transient and the risk associated with such a transient, while small, should not be incurred. Based on this analysis and the avoided risk of an unnecessary plant transient, a loss of load from below the P-9 setpoint and a 50% load reduction from 100% EPU power to verify proper operation of the plant and automatic control systems is not required in the PBNP EPU Power Ascension Test Plan. Further, load step power changes and load ramp testing is not necessary and will not be performed for EPU conditions.

Manual Turbine Trip from 100% Power Test

The manual turbine trip from 100% power is a test to demonstrate that the control systems act together to maintain NSSS parameters within design limits post-trip and to demonstrate MSSVs do not open. In particular, the test demonstrates that the rod control, steam dump and pressurizer pressure and level control systems act together to control the NSSS response to within design limits and prevent opening of MSSVs. An analysis of a turbine trip from 100% EPU power was performed using the LOFTRAN code as described in LR Section 2.4.2, Plant Operability. This analysis demonstrates that the PBNP plant response to a turbine trip at full power EPU conditions results in acceptable response of pressurizer level and pressure, and MSSVs do not open.

There are no major hardware modifications planned for NSSS components as part of the EPU that would affect the plant transient response. Since the NSSS control system functional design and hardware are not impacted and the analyzed turbine trip from 100% EPU power Condition II operating transient shows acceptable stability, setpoints, and margin to ESF actuation, the NSSS control systems are acceptable for operation at full power EPU conditions. A reactor trip, or the potential for a reactor trip, from high power level results in an unnecessary plant transient and the risk associated with such a transient, while small, should not be incurred. Based on this analysis and the avoided risk of an unnecessary plant transient, a manual turbine trip from 100% EPU power to verify proper operation of the plant and automatic control systems is not required in the PBNP EPU Power Ascension Test Plan.

Natural Circulation Test

The purpose of the natural circulation test is to demonstrate the capability of natural circulation to remove core decay heat while maintaining NSSS parameters within design limits. The test was performed as part of original startup testing at 2% power and demonstrated that natural circulation flows were adequate to remove heat and maintain NSSS parameters in an acceptable range.

To evaluate the natural circulation capability for the PBNP EPU, the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG) methodology is used to estimate flow rates and core delta temperatures using core hydraulic resistance coefficients. These equations are evaluated for several decay heat assumptions (1, 2, 3, and 4%) over a range of temperature conditions. This analysis of natural circulation cooldown to residual heat removal (RHR) cut-in conditions is described in more detail in LR Section 2.8.7.2, Natural Circulation Cooldown.

In addition, the atmospheric dump valve (ADV) capacities are estimated as function of steam generator secondary pressure that is correlated with primary system saturated temperature. After 4 hours at hot standby conditions, the plant is assumed to cool down to the RHR cut-in conditions at the maximum Emergency Operating Procedure (EOP) rate (25°F/hour).

There is close agreement between the hydraulic resistance coefficients for the Diablo Canyon and PBNP plants at the uprated conditions and the loop flow ratios are in good agreement. The calculated loop delta temperatures show the same trends and slightly higher scaled values compared to the FSAR reported measured values. The natural circulation flow rate shows expected behavior - decreases as the decay heat decreases at a constant temperature and a decrease with temperature at a constant value of decay heat. The loop delta temperature shows expected behavior - decreases as the decay heat decreases at a constant core average temperature and increases as the core average temperature decreases at a constant value of decay heat.

For the following reasons, the PBNP EPU will not adversely impact the natural circulation cooldown capability of the plant:

- No major hardware modifications to NSSS components that could affect loop flow resistance or steam generator heat transfer are part of the EPU scope.
- Acceptable results are found for natural circulation cooling during the hot standby period for realistic residual heat rates as high as 3% of 1811 MWt. The core outlet temperatures calculated for this case (604.5°F) are bounded by those specified for full power operation for the high $T_{avg}$ cases (611.8°F) (PCWG Cases 3 and 4, LR Section 1.1, Nuclear Steam Supply System Parameters, Table 1-1).
- The calculated loop delta temperatures are scaled and compared to the FSAR measured values. The scaled, calculated values show the same trends as the original measurements and are slightly larger than measured, due to several conservative assumptions in the calculations. One of the conservative assumptions is that the hydraulic resistance for the reactor coolant pump (RCP) is based upon a locked-rotor K value.
- The atmospheric dump valves (ADVs) at the uprated conditions are adequate to achieve cooldown to the RHR entry point in an acceptable time period. RHR cut-in conditions can be achieved in approximately 14 hours at the maximum rate specified in Emergency Operating Procedures, which includes 4 hours in hot standby conditions.

### 2.12.1.3 Conclusions

PBNP has reviewed the EPU test program, including plans for the initial approach to the proposed maximum licensed thermal power level and the test program's conformance with applicable regulations. PBNP concludes that the proposed EPU test program provides adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the proposed EPU, or modified to support the proposed EPU, will perform satisfactorily in service. Further, PBNP finds that there is reasonable assurance that the EPU testing program satisfies the requirements of 10 CFR 50, Appendix B, Criterion XI. Therefore, PBNP finds the proposed EPU test program acceptable.

### 2.12.1.4 References


1. WCAP-7907 P-A (LOFTRAN Code Description), dated April 1984

## Table 2.12-1
## PBNP Extended Power Uprate Power Ascension Test Plan

| Test/Modification | Test Description | Prior To Startup | Rated Thermal Power, % of 1800 MWt (Allowance +0%, -5%) | | | | | | | | | | | | | | | | | (Allowance +0%, -1%) | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | 0 | 5 | 10 | 15 | 20 | 25 | 30 | 35 | 40 | 45 | 50 | 55 | 60 | 65 | 70 | 75 | 80 | 85 | 88 | 91 | 94 | 97 | 100 |
| Nuclear Steam Supply System Data Record | Data Collection | | X | | | | X | | X | | | | X | | | | | X | | X | X | X | X | X | X |
| Balance of Plant Data Record | Data Collection | | X | | | | X | | X | | | | X | | | | | X | | X | X | X | X | X | X |
| Transient Data Record | Data Collection | | | | | | X | | X | | | | X | | | | | X | | X | X | X | X | X | X |
| Nuclear Design Check Tests | Low Power Physics Testing (Item 32 Table 2.12-2) | | | X | | | | | | | | | | | | | | | | | | | | | |
| Power Distribution Monitoring | Performing Core Flux Maps | | | | | | | | X | | | | X | | | | | X | | | | | X | | |
| Core Power Determination | Plant Calorimetric (Item 34 Table 2.12-2) | | | | | | | | | | | | | | | | | | | X | | | X | X | X |
| RCS Flow Measurement | Verification of RCS Flow (Item 1 Table 2.12-2) | | | | | | | | | | | | | | | | | | | | | | X | | |
| Leading Edge Flow Meter Calibration Checks | Verification of Calibration of LEFM | | | | | | | | | | | | | | | | | | | X | | | X | X | X |
| Vibration Monitoring | Monitor vibration in Plant Piping and Rotating Equipment | | X | | | | | | | | | | | | | | | | | X | X | X | X | X | X |
| Plant Radiation Surveys | Verify Expected Dose Rates | | | | | | | | | | | | | | | | | | | X | | | | | X |

## Table 2.12-1
## PBNP Extended Power Uprate Power Ascension Test Plan

| Test/Modification | Test Description | Prior To Startup | Rated Thermal Power, % of 1800 MWt (Allowance +0%, -5%) | | | | | | | | | | | | | | | | | | (Allowance +0%, -1%) | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | 0 | 5 | 10 | 15 | 20 | 25 | 30 | 35 | 40 | 45 | 50 | 55 | 60 | 65 | 70 | 75 | 80 | 85 | 88 | 91 | 94 | 97 | 100 |
| Plant Temperature Surveys | Verify Expected Temperatures | | | | | | | | | | | | | | | | | | | X | | | | | X |
| Moisture Carryover Test | Verification MCO 0.25 percent | | | | | | | | | | | | | | | | | | | | | | | | X |

Note:
1. The 85% plateau corresponds to the current licensed power level, (1540 MWt, approximately 85.6% of EPU power).

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 1 | Reactor Coolant System | Yes | To verify that all instrumentation and control functions of the system were operating properly and that system flows were correct. | The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Specifically, the flow rate though the reactor coolant system will change by only a negligible amount as a result of EPU. System instrumentation will be checked out as part of the plant surveillance program required for startup. Measurement of reactor power will be performed a power levels identified in Table 2.12-1. At the 94% plateau, the reactor power measurement will be used as input to the determination of RCS flow. The test is performed routinely to satisfy Tech Spec Surveillance Requirements. |
| 2 | Component Cooling | No | To verify component cooling flow to components served by the system and proper operations of valves, instrumentation and alarms associated with the system. | The component cooling system has been assessed and determined to be adequate to support uprate. However, selected component cooling parameters will be monitored during escalation to power. |
| 3 | Residual Heat Removal System Test | No | To verify proper operation of valves, instrumentation and alarms associated with the system and the ability of the system to cool the plant from 350°F to 140°F in 20 hours. | RHR system capabilities are adequate for the power uprate condition and that the power uprate has no adverse affect on this system. There are no modifications planned to the RHR system for EPU. Therefore, this test is not required to be performed at the uprated power conditions. Additionally, the operability of this system is verified by regular surveillance testing. |

## Table 2.12-2
## EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 4 | Spent Fuel Pool Cooling | No | To verify proper operation of valves, instrumentation and alarms associated with the system and proper flow paths for cooling. | No modifications have been performed on Spent Fuel Pool Cooling, therefore, this test is not required to be performed at the uprated power conditions. Spent fuel pool conditions are routinely monitored during plant operation. |
| 5 | Chemical and Volume Control System | No | To verify that the system performed the following functions: maintain reactor coolant system water inventory, borate and dilute the reactor coolant system, supply reactor coolant pump seal water, maintain primary water chemistry within acceptable limits. | This test was performed during Hot Functional Testing, prior to fuel load. No modifications were made to this system, and there will be only small changes in the reactor coolant system parameters. Therefore, this test is not required to be performed at the uprated power conditions. However, selected parameters will be monitored during the power ascension testing. |
| 6 | Sampling System | No | To verify that a specified quantity of representative fluid and gases could be obtained safely at design conditions from each sampling point. | This test was performed during Hot Functional Testing, prior to fuel load. Primary and Secondary samples will be taken and analyzed at full power as a matter of normal plant operations. |
| 7 | Waste Disposal System | No | To demonstrate that the system was capable of processing all radioactive liquids, gases and solids associated with plant operation. | The waste disposal system is not impacted by power uprate. Therefore, this test is not required to be performed at the uprated power conditions. |

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 8 | Safety Injection Test | No | To verify proper response of the system to actuating signals in regards to pump, valve, instrumentation and alarms associated with system. | The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Further, operability of the SI system is verified by standard surveillance testing. Therefore, this test is not required to be performed at the uprated power conditions. |
| 9 | Fuel Handling | No | To demonstrate that the system was capable of handling fuel in all circumstances which would occur from receipt of fuel to return of fuel in a safe and orderly manner. | The fuel handling system is not impacted by power uprate. Therefore, this test is not required to be performed at the uprated power conditions. Note that the fuel handling system is used extensively during refueling activities and is inherently undergoing thorough testing. |
| 10 | Reactor Protection System | No | To verify the reactor tripping circuitry by operationally checking the analog system tripping and the A and B logic trains. | The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the logic of the Reactor Trip System will not be changed as a part of this EPU and the test does not need to be repeated since the initial testing had satisfactory results. New reactor trip setpoints for EPU will be verified by instrument calibration tests. Additionally, the operation of these systems is verified by regular surveillance testing. |

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 11 | Rod Control System | No | To verify the rod control system satisfactorily performed the required stepping operations for each individual rod under both cold and hot shutdown conditions and to determine the rod drop time for each full length RCCA, and to check out the part-length rod drive system. | The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the parameters of concern for this test are not altered by EPU, and the rod control system has performed its intended function during all phases of plant operation. The operation of these systems is verified by regular surveillance testing. |
| 12 | Rod Position Indication System Test | No | To verify the rod position indication system satisfactorily performed the required indication and control for each individual rod under hot shutdown conditions. | The rod position indication system is not impacted by power uprate. Therefore, this test is not required to be performed at the uprated power conditions. The operation of this system is inherently tested during refueling and regular physics testing. |
| 13 | Feedwater Control System | No | To demonstrate that the steam generator water level could be controlled in the manual and the automatic mode of operation and to ensure that all alarms and trips were functioning properly. | The feedwater system and controls will be modified to support power uprate. Proper operation of controls will be verified through post-modification testing. Selected system parameters will be monitored during power escalation. Finally, the planned load swing tests will dynamically test the FW control system. See Section 2.12.1.2.3. |
| 14 | Steam Dump Control System | No | To verify proper settings of the steam dump control system and the capability of the steam dump system to reduce the transient conditions imposed as a result of a load cutback or rejection up to 50% without a reactor trip. | No changes to the steam dump valves or setpoints are being made for EPU conditions. The system will not be dynamically tested via large load rejection testing; see Section 2.12.1.2.6 where justification is provided for not performing the 50% load rejection test. |

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 15 | Nuclear Instrumentation Test | No | To verify the proper operation of the Nuclear Instrumentation System. | The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, this test provided a functional demonstration of the system only. Additionally, the operation of these systems is verified by regular surveillance testing. |
| 16 | Radiation Monitoring System Operational Test | No | To verify that all channels were operable and alarm and recording functions were responding properly. | The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Additionally, the operation of these systems is verified by regular surveillance testing. |
| 17 | In-Core Instrumentation System | No | To perform checkout and demonstration of the in-core thermocouple system and the in-core flux mapping system. | The power uprate has no adverse affect on the system and does not invalidate the test as originally performed. The In-Core Detector System is used during normal plant operation and has proven itself to be reliable. Therefore, these tests are not required to be performed at the uprated power conditions. Specifically, the in-core instrumentation and thermocouple readouts are not adversely impacted by the uprate, and the operation of these systems is verified by regular surveillance testing. |

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 18 | Service Water System | No | To verify that the system would supply the required water flow through all equipment supplied with service water and that all instrumentation and controls functioned as designed. | The service water system has been assessed and determined to be adequate to support uprate. It is noted however, that selected service water parameters will be monitored during escalation to power. |
| 19 | Fire Protection System | No | To verify proper operation of the system and to check all automatic functions. | The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. |
| 20 | Circulating Water System | No | To verify proper operation of pumps, valves and control circuitry; proper priming of the system, and proper flow through the condensers and the condensate cooler. | The circulating water system was assessed and found to be adequate to support uprate. Therefore, this test is not required to be performed at the uprated power conditions. Selected system parameters will be monitored during power escalation. |

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 21 | Instrument and Service Air System | No | To verify:<br>a. the proper operation of all compressors to design specifications,<br>b. the manual and automatic operation of controls at design setpoints,<br>c. design air dryer cycle time and moisture content of discharge air,<br>d. proper air pressure to each instrument and equipment served by the system. | Modifications to the air systems as a result of EPU modifications will be performed as part of the post modification testing. There are no required additional tests to support plant uprate. Therefore, this test is not required to be performed at the uprated power conditions. |
| 22 | Reactor Containment Air Circulating System | No | To verify the proper operation of:<br>a. all fans, filters, heating and cooling coils,<br>b. automatic and manual controls to maintain containment atmosphere within design specifications,<br>c. proper operation of recirculation fans and coolers on a safety injection signal,<br>d. purge valve isolation,<br>e. all interlocks and alarms. | The power uprate has no adverse affect on the system and does not invalidate the test as originally performed. The system is adequate to handle the slight increase in containment heat load. Therefore, this test is not required to be performed at the uprated power conditions. Note however, that selected system parameters will be monitored during power escalation. |

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 23 | Feedwater and Condensate System | No | To verify pump, valve, and control operability and set-points. Functional testing was performed when a steam supply was available. | The feedwater system and condensate systems will be modified to support power uprate. New equipment (Condensate and FW pumps, FW heaters) performance will be monitored and system adequacy will be verified through post-modification testing. Further, selected system parameters will be monitored during power escalation. |
| 24 | Control Room Ventilation System | No | To demonstrate the control room ventilation system could perform its designed function during normal plant operations and during postaccident plant conditions by checking out each mode of operation. | The EPU did not modify the ventilation system and the testing/balancing that was performed during startup is still valid; therefore, testing of the ventilation system will not be performed. Monitoring of general area temperatures, particularly those areas where new equipment is installed, will be performed as part of the power ascension test procedure to confirm that the ventilation system continues to perform its intended function. |
| 25 | Emergency Diesel Generator Test<br>This test verified the air capacity needed to crank the engines for 45 seconds. It also verified that the diesel could be placed on line within 10 seconds. | No | To assure that the emergency diesel-generators were installed in accordance with the design specifications and operated as described in the functional description to satisfactorily accept the safeguard system load upon failure of the normal power supply. | The power uprate has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the diesel start time, load time, and capacity were validated by this test. These requirements do not change as a result of the power uprate. Additionally, the operation of these systems is verified by regular surveillance testing. |

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 26 | Switchgear System | No | To verify that the electrical, auxiliary, and safeguard systems were installed and operated in accordance with accepted electrical standard and design and thereby provided reliable power to auxiliaries required during any normal or emergency mode of plant operation. | No modifications to plant switchgear were required to support power uprate. Therefore, this test is not required to be performed at the uprated power conditions. |
| 27 | Primary System Safety Valves Tests | No | To ascertain the popping and reseat pressure settings of the valves and establish that zero leakage conditions existed across the seating face. | The power uprate has no adverse affect on safety valves and does not invalidate the test as originally performed. The Main Steam Safety Valve setpoints are being revised and the valve setpoint will be tested as part of the modification implementation. In addition new FW Isolation Valves are being installed and will be tested as part of the post-mod test. Safety Valves are routinely tested as required by the ASME Code but not during power ascension testing. |
| 28 | Reactor Containment High Pressure Test and Leakage Test | No | To verify the structural integrity and leak tightness of containment. | This test is performed at intervals directed by Technical Specifications. This test does not have to be performed for uprate because the EPU did not modify the containment structure or penetrations in any way. |
| 29 | Cold Hydrostatic Tests | No | To verify the structural integrity and leak tightness of the particular system. | Hydrostatic testing of modified systems will be performed during the post modification testing as required PBNP station requirements. No specific EPU power ascension testing is therefore required. |

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 30 | RCC Unit Drop Tests | No | To measure the drop times of all RCC units from loss of coil voltage to dashpot entry at cold and hot conditions with full flow. Selected rods will be dropped at no flow conditions. | No modifications for the control rod system are required for EPU; therefore this test is not required to be re-performed at the EPU condition. Rod drop testing is performed as part of normal low power physics testing during refueling activities. |
| 31 | Thermocouple/RTD Inter-calibration<br>This procedure was used to determine the isothermal corrections for reactor coolant resistance temperature detectors and in-core thermocouples. | No | To verify RTD calibration data and to determine in-place isothermal correction constants for all core exit thermocouples. | The EPU will marginally raise the reactor coolant temperature. This testing and cross- calibration is performed as part of normal reactor start-up. |
| | Nuclear Design Check Tests | Yes | To verify that the nuclear design predictions for endpoint boron concentrations, isothermal temperature coefficients, RCC bank differential and integral worths and power distributions are valid. | Nuclear checks are performed as directed by the Core Operating License Report following each refueling outage. Other core parameters are verified to be in specification before exceeding 50% power as required by Technical Specifications. |

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 33 | Plant Trip | No | To verify reactor control performance control and steam dump performance. | This test was originally performed at 30% and 100% power. The power uprate does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. See Section 2.12.1.2.6 for additional justification for not performing this test. |
| 34 | Plant Calormetric and Power Range Instrumentation Calibration | Yes | During static and/or transient conditions at approximately 40%, 70%, 90% and 100%.<br>To calibrate power range channels such that total core thermal output is indicated and that the detectors indicated the relationship between incore and excore axial offsets and quadrant tilts. | Nuclear instrumentation calibration is performed at various power levels as part of normal reactor start-up. The flow confirmation test is not impacted by EPU, but a calorimetric flow test will be performed at 85% and 100% EPU power. |
| 35 | Load Swing and Load Reduction Test | No | a. ±10% at approximately 25%, 60% and 100% of rated power<br>b. Load reduction of approximately 50% from 55% and 100% power level<br>c. Ramp load increase and decrease between 40% and 90% at the rate of 5%/minute. | This test was originally performed at several power levels to verify the adequacy of various plant systems to respond to load swings. See Section 2.12.1.2.6 for justification for not performing the load reduction and ramp transient tests. |

**Table 2.12-2**
**EPU Test Plan and Comparison of Proposed EPU Tests to Original Startup Tests**

| Item No. | Test Description<br>FSAR Table 13.2-1 | Test Plan For EPU (yes/no) | Initial Startup Test Objective | EPU Test Basis<br>The scope of EPU planned testing is described in this column |
|---|---|---|---|---|
| 36 | Dynamic RCC Drop Test | No | To verify automatic detection of dropped rod by bottom and power range detector indication for selected rods. A minimum of one drop be accompanied with turbine runback and automatic rod withdrawal stop. | The dropped rod recovery procedure was proven adequate and in subsequent testing, the turbine runback controller performed as designed.<br>This system has been fully tested and found to be satisfactory and the EPU will not affect this system so testing again is not necessary. |
| 37 | Static RCC Insertion and Drop Tests | No | To verify that a single RCC unit when misaligned with the control bank can be detected by individual rod position indication or by incore instrumentation if required. To determine the effect of a single full inserted RCC unit on core reactivity and core power distribution. | The dropped rod recovery procedure was proven adequate and in subsequent testing, the turbine runback controller performed as designed.<br>This system has been fully tested and found to be satisfactory and the EPU will not affect this system so testing again is not necessary. |
| 38 | Radiation Shielding Effectiveness Test | Yes | a. 10-8 - 10-7 amps<br>b. 1 - 3%<br>c. 30 - 40%<br>d. 100%<br>Measure neutron and gamma shielding effectiveness in the containment. | Radiation shielding measurements performed at lower power levels are not invalidated by EPU. However, plant surveys, including radiation shielding measurements will be performed at the power levels shown in Table 2.12-1, and survey maps updated as necessary. |