AUG 20 2013

L-2013-227
10 CFR 50.36

FPL.

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, D.C. 20555-0001

Re: Turkey Point Unit 4
Docket No. 50-251
Renewed Facility Operating License No. DPR-41
Extended Power Uprate Cycle 27 Startup Report-Supplement 1

References:

(1) J. Page (NRC) to M. Nazar (FPL), Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Extended Power Uprate (TAC Nos. ME4907 and ME4908), Accession No. ML11293A365, June 15, 2012.
(2) M. Kiley (FPL) to U. S. Nuclear Regulatory Commission (NRC) (L-2013-218), Renewed Facility Operating License No. DPR-41 Extended Power Uprate Cycle 27 Startup Report, July 16, 2013.

Pursuant to Turkey Point Unit 4 Technical Specification 6.9.1.1, Florida Power & Light Company (FPL) is providing supplement 1 to the Unit 4 Cycle 27 Startup Report submitted on July 16, 2013 via Reference 2. The Startup Report and supplemental report are required due to the implementation of Unit 4 Amendment No. 245 for the Extended Power Uprate (EPU) Project that was issued via Reference 1.

At the time that the initial Startup Report was issued, the remaining 10% transient ramp tests from 100% rated thermal power (RTP) to 90% RTP, and then from 90% RTP back to 100% RTP had not been completed. In addition steady state data between 89% and 100% power had not been fully summarized in time for the report. The final 10% ramp tests have been completed and the results of all remaining power ascension testing for Unit 4 are being submitted in the attached final report.

Sincerely,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachment

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Senior Resident Inspector, Turkey Point Nuclear Plant

IE26
NRR

TURKEY POINT UNIT 4 CYCLE 27
SUPPLEMENT 1
EXTENDED POWER UPRATE STARTUP REPORT

Table of Contents

I. Introduction

The purpose of this Startup Report is to provide a summary description of the plant startup and power ascension testing performed at Turkey Point Unit 4 following Cycle 27 refueling which implemented the Extended Power Uprate (EPU) project. The EPU License Amendment Request (LAR) 205 was approved on June 15, 2012 through license amendment 245 [Reference 1]. The amendment increased the authorized maximum steady-state reactor core power from 2300 megawatts thermal (MWt) to 2644 MWt. This Unit 4 Cycle 27 Startup Report is being submitted in accordance with Turkey Point Technical Specification 6.9.1.1, items (2) and (4).

Technical Specification 6.9.1.1, Startup Report, states that, "a summary report of plant startup and power escalation testing shall be submitted following: ... (2) amendment to the license involving a planned increase in power level and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions of characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications."

All required testing per the Startup Test Program has been completed for Turkey Point Unit 4 Cycle 27.

Technical Specification 6.9.1.1, Startup Report, states "If the Startup Report does not cover all three events (i.e. initial criticality, completion of the Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted every 3 months until all three events have been complete." In accordance with the Technical Specifications, this is supplement 1 to the initial report and completes all required reporting for Turkey Point Unit 4 Cycle 27 startup.

The plant startup and power ascension testing verifies that key EPU core and plant parameters are operating as predicted. The major parts of this testing program include:

1) Cycle 27 core design summary,
2) Low power physics testing, and
3) Power ascension testing.

The test data collected during EPU power ascension summarized in this report concludes that all major systems, structures, and components (SSCs) performed as predicted and there was no adverse impact to the performance of the unit. All startup testing results related to core performance and power ascension data review between 20% and 87% power were found acceptable and previously reported in Reference 4. Therefore, they are not repeated in this report. Copies of the completed EPU power ascension test procedures are available on site for review.

II. Power Ascension Test Program

The EPU power ascension test program consisted 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. During the EPU start-up, power was 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 were used until the previously licensed full power condition (2300 MWt) was attained (approximately 87% of the EPU full power level of 2644 MWt). Above 2300 MWt, 3% intervals between test plateaus were established for data acquisition and evaluation. A summary of the power ascension test plan from LAR No. 205 [Reference 2] is provided in Tables 2.12-1 and 2.12-2 below.

Table 2.12-1
PTN Extended Power Uprate Power Ascension Test Plan

Test / Modification	Test Description	Prior To Startup	Rated Thermal Power - % of 2644 MWt (Allowance +0%, -5%)																	(Allowance +0%, -1%)					
			0	5	10	15	20	25	30	40	45	50	55	60	65	70	75	80	87	89	92	95	98	100
Nuclear Steam Supply System (NSSS) Data Record	Data Collection						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
Transient Data Record[6]	Data Collection						X		X			X							X	X	X	X	X	X
Core Map	Power Distribution and COLR Parameters								X[5]			X					X[1]		X	X[1]	X[1]	X[1]	X[1]	X
NSSS Calorimetric and Power Range Channel Adjustment	Verify Thermal Power and Adjust Nuclear Instrumentation						X		X			X					X		X	X	X	X	X	X
Reactor Coolant System (RCS) Flow Measurement	RCS Flow Calorimetric																							X

Table 2.12-1

PTN Extended Power Uprate Power Ascension Test Plan

Test / Modification	**Test Description**	**Prior To Startup**	**Rated Thermal Power - % of 2644 MWt (Allowance +0%, -5%)**																	**(Allowance +0%, -1%)**				
			0	**5**	**10**	**15**	**20**	**25**	**30**	**40**	**45**	**50**	**55**	**60**	**65**	**70**	**75**	**80**	**87**	**89**	**92**	**95**	**98**	**100**
Incore-Excore Axial Offset Calibrations	Calibrate Excore Instrumentation to Incore Axial Offset									**X**[2]									**X**[3]					**X**[4]
Load Changes	10% Ramp to Verify System Response								**X**															**X**[7]
Turbine Trip	OST Turbine to Verify System Response			**X**																				
Turbine Stop Valve, Governor Valve, and Intercept Valve Testing.	Standard turbine valve tests w/post-modification tests									**X**[6]														
Steam Generator Level Feedwater Flow Response Testing	Manually inserted level setpoint step-changes in the steam generator.								**X**										**X**			**X**		

Table 2.12-1

PTN Extended Power Uprate Power Ascension Test Plan

Test / Modification	**Test Description**	**Prior To Startup**	**Rated Thermal Power - % of 2644 MWt (Allowance +0%, -5%)**																	**(Allowance +0%, -1%)**					
			0	**5**	**10**	**15**	**20**	**25**	**30**	**40**	**45**	**50**	**55**	**60**	**65**	**70**	**75**	**80**	**87**	**89**	**92**	**95**	**98**	**100**
Vibration Monitoring	Monitor Vibration in Plant Piping and Supports	**X**																	**X**	**X**	**X**	**X**	**X**	**X**
Thermal Expansion	Monitor Thermal Expansion in Plant Piping	**X**																	**X**	**X**	**X**	**X**	**X**	**X**
Plant Radiation Surveys	Verify Expected Dose Rates																		**X**					**X**
Plant Temperature Surveys	Verify Expected Temperatures																		**X**					**X**

NOTES:

(1) If required
(2) Incore flux map for data acquisition will be performed at 50% of 2644MWt or when annunciator B2/2 or B2/3 alarms, whichever comes first.
(3) Incore flux map for data acquisition will be performed at approximately 87% of 2644 MWt, if required.
(4) At steady state equilibrium Xenon conditions
(5) Not in LAR 205, performed as part of normal startup procedures
(6) Test moved from 35% to 50% to ensure that ESFAS high steam flow and AMSAC arming setpoint are not adversely affected by steam flow imbalance.
(7) Down

Table 2.12-2
Large Plant Transient Tests in Turkey Point EPU Power Ascension Test Plan

Proposed Test	Description	Expectation
Turbine Overspeed Trip from 5% EPU Power	The turbine will be, with the reactor at approximately 5% power, automatically tripped as speed exceeds the electronic overspeed trip setpoint.	This test will verify proper operation of the overspeed mechanism for the new EHC turbine control, and proper operation of the new turbine control valves.
10% Ramp Load Change at new 30% and 100% EPU Power	Ramp load change limited by station license conditions and fuel pre-conditioning considerations.	These ramps will test NSSS and BOP control system operation, to ensure that no unanticipated aggregate effects have been produced by interaction of the plant modifications.
Turbine Stop Valve, Governor Valve, and Intercept Valve Testing at 35% EPU Power	Standard turbine valve testing augmented by post-modification tests associated with EHC Turbine control and Governor Valve Replacement.	Validate dynamic performance of new governor valve design to ensure adequate transient response. Verify acceptable dynamic performance of the new HP turbine rotor during changes in individual arc steam flows.
Steam Generator (SG) Level / Feedwater Flow Dynamic testing at 30%, 87% and 95% EPU Power	Verify response to manually inserted level setpoint step-change of 5% in the steam generator. Both up-going and down-going setpoint changes of different magnitudes will be inserted.	Verify SG level control system response and acceptability of over-shoot, damping and steady-state limit cycling at the new licensed power level. Verify acceptable operation of the feedwater control system.

FPL provided specific acceptance criteria in response to NRC request for information [Reference 3] for the 10% ramp load change list in Table 2.12-2 above as follows:

- RCS average temperature, pressurizer pressure, and pressurizer water level will be controlled to the programmed values.
- Steam generator water level will demonstrate good feedwater level control and maintain acceptable margin to the trip level setpoint.
- Nuclear power peak overshoot/undershoot should be less than 3 percent reactor thermal power.

- Steam generator water level should return to programmed level setpoint within ±2 percent narrow range with dampening oscillations within 15 to 20 minutes.

Prior to exceeding the previous licensed core thermal power of 2300 MWt, the data gathered at the pre-determined power plateaus, as well as observations of the slow, but dynamic power increase between the power plateaus, allowed verification of the performance of the EPU modifications. The steady-state data collected at approximately 87% power was especially significant because this test plateau corresponded to the previous 100% core power level of 2300 MWt. Data collected at this plateau formed the basis for comparison of data collected at higher plateaus.

Once testing was completed at the 2300 MWt plateau, power was slowly and deliberately increased through four 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 were monitored, documented and evaluated against predetermined acceptance criteria and expected values. The acceptance criteria for the power ascension test plan were established as discussed in Regulatory Guide (RG) 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants and NUREG 0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, Section 14.2.1. Criteria were provided against which the success or failure of the test was judged. In some cases, the criteria were qualitative where applicable, quantitative criteria had appropriate tolerances.

Specific acceptance criteria and expected values were established and incorporated into the power ascension test procedures. 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 judged to be safe based upon prior testing. Resolution of the issue that resulted in exceeding the Level 1 criterion must be resolved by equipment changes or through engineering evaluation, as appropriate. 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 exceeding the Level 2 criterion may continue in parallel with the power escalation. These investigations would be handled by existing plant processes and procedures.

Following each increase in power level, test data was evaluated against its performance acceptance criteria (Level 1 and 2 and expected values, i.e., prediction targets for power level). If the test data satisfied the acceptance criteria, then system and component performance were considered to have complied with their design requirements. Predicted values are used for optimizing SSC performance only and are not acceptance criteria.

In addition to the steady-state parameter data gathered and evaluated at each test condition, the dynamic parameter response data gathered during the ascension between test plateaus was also evaluated for overall stability of the plant.

Hydraulic interactions between the new main feedwater pumps and the steam generator flow control valves, as well as the impact of the higher main feedwater flow, were monitored and evaluated. Individual control systems, such as steam generator level control and feedwater heater drain level control, were optimized for the new EPU conditions, as required.

Vibration Monitoring

A piping and equipment vibration monitoring program, including plant walkdowns and monitoring of plant equipment, was established to ensure that any steady-state flow induced piping vibrations following EPU implementation were not detrimental to the plant, piping, pipe supports, or connected equipment.

The predominant way of assessing piping and equipment vibrations was to monitor the piping during the plant heat-up and power ascension. The methodology used for monitoring and evaluating vibration was in accordance with ASME OM-S/G-2007, Standards and Guides for Operation and Maintenance of Nuclear Power Plants, Part 3, Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems.

The scope of the piping and equipment vibration monitoring program included accessible piping that experienced an increase in process flow rates. Branch lines attached to this piping that experienced increased process flows were also monitored as operating experience has shown that branch lines are susceptible to vibration-induced damage. The scope of the program included the following systems:

- Main steam (outside of containment) including Reheater Inlet,
- Main Steam modified supports (inside containment),
- Feedwater (outside of containment),
- Condensate,
- Extraction Steam,
- Heater Drains,
- Moisture Separator Drains, and
- Turbine Gland Steam and Drains.

III. Power Ascension Test Program Results

NSSS Data Collection

The Turkey Point Unit 4 nuclear steam supply system (NSSS) significant parameters were observed at the 20%, 30%, 50%, 75%, 87%, 89%, 92%, 95%, 98%, and 100% EPU power plateaus. During power ascension between 89% and 100%, the NSSS significant parameter values at the various power plateaus met all Level 1 and Level 2 criteria with the exception of main steam flow.

Similar to Unit 3, during escalation to the 92% power plateau, momentary single channel High steam flow alarms were received for the B SG which exceeded Level 1 test criteria. The High flow alarms were caused by signal spikes from flow turbulence in the main steam (MS) and feedwater (FW) lines. Prior to reaching 92% power, installation of lag functions in the MS and FW flow circuits had started based on Unit 3 experience but had not been completed. The alarming High steam flow circuit was modified with the lag function before increasing power above 92%. All remaining MS and FW flow circuits had lag functions installed prior to exceeding 98% power. Post modification testing showed no spiking in the MS flow circuits and at least a 50% reduction in FW flow signal amplitude. The circuit modifications provided adequate margin to the trip set point for FW and MS flow signals. At the 100% power plateau, none of the NSSS Level 1 and Level 2 criteria were exceeded.

Below provides a brief summary of major control parameters at 100% power. In addition, Table 1 provides a summary of the NSSS significant parameters between 89% and 100% power plateaus. Reference 4 previously reported data between 20% and 87% power.

- RCS average temperatures – As can be seen from Table 1, the maximum measured average RCS temperature at 100% EPU power is 579.9°F which is below the EPU full power limit of 581.5°F.

- Pressurizer pressure – remained at 2235 psia +/- 5% within normal operating band throughout the power ascension and transient test.

- Pressurizer level – The pressurizer level program changes from 22.5% at RCS average temperature 547 °F to 56.9% at full load RCS average temperature of 580°F. Pressurizer level was maintained on program within normal control system tolerances throughout the power escalation. At 100% power, average pressurizer level is 56.3%.

- Containment temperature – temperature ranged from 105.3°F to 110.0°F throughout the power ascension, well below the 125 °F limit.

- Steam generator header pressure – ranged between 783 psig at 89% EPU power and 799 psig at 100% EPU power.

- Steam generator level – remained constant at 50% narrow range scale within normal control system tolerances throughout the power ascension.

The NSSS data collected during the EPU power ascension testing is too extensive to include in this summary report. The completed test procedures and all BOP data collected at the 20%, 30%, 50%, 75%, 87%, 89%, 92%, 95%, 98%, and 100% EPU power plateaus are available for review on-site, if required.

Balance of Plant (BOP) Data Collection

The Turkey Point Unit 4 BOP significant parameters were observed at the 20%, 30%, 50%, 75%, 87%, 89%, 92%, 95%, and 100% EPU power plateaus. As the majority of the EPU hardware changes were made to BOP equipment, extensive monitoring of the secondary side was performed during the EPU power ascension. Major systems and components monitored included:

- High pressure turbine, low pressure turbine, main generator and exciter vibration,
- High pressure turbine, low pressure turbine, main generator and exciter bearing temperatures,
- High and low pressure turbine steam pressure and temperature,
- Moisture separator reheater (MSR) pressure and temperature,
- Turbine digital controls,
- Main generator gas temperatures,
- Turbine cooling water system performance,
- Condensate, main feedwater, and heater drain system pressure and temperature,
- Condensate, main feedwater, and heater drain pump performance,
- Feedwater heater performance,
- Heater drain valve performance,
- Main condenser performance,
- Main transformer performance,
- Isolated phase bus cooling performance, and
- Main generator electric output.

A portion of the BOP data was obtained through walkdowns at each plateau. The purpose of the walkdowns was to visually observe operation of accessible components, not available on the plant process computer, at each plateau during the power ascension. Multiple test personnel were used to accomplish the walkdowns and the test personnel discussed all observations and findings prior to power escalation. The corrective action program was utilized to document any walkdown findings. Several instruments were found either out of calibration or required repair. Once the instruments were repaired or alternative instruments provided, accurate data was obtained.

None of the BOP parameters in the power ascension test program exceed Level 1 or Level 2 criteria with the exception of the Heater Drain Tank (HDT) discharge valve position. HDT 3B normal control valve CV-4-1510A, was found to be approximately 91% open which exceeds the Level 2 criteria of < 85%. Control valve Level 2 criteria of less than 85% open is established to allow sufficient valve capacity for normal operation transients. In the case of the HDT level control system, a separate high level dump valve CV-4-1510B is provided for high level conditions caused by normal operation transients. Therefore, having CV-4-1510A at 91% open was found acceptable.

The BOP data collected during the EPU power ascension testing is too extensive to be included in this summary report; however, Table 1 provides a summary of major system parameters monitored between 89% and 100% power. Reference 4 previously reported data between 20% and 87% power. The completed test procedures and all BOP data collected at the 20%, 30%, 50%, 75%, 87%, 89%, 92%, 95%, 98%, and 100% EPU power plateaus are available for review on-site, if required.

Vibration Monitoring

The Turkey Point Unit 4 piping and equipment within the scope of the EPU vibration monitoring program were observed at several different plant operating conditions, namely the 30%, 50%, 87%, 89%, 92%, 95%, 98%, and 100% EPU power plateaus. The first observations were conducted prior to the shutdown in which the EPU modifications were implemented. Data from these observations was used to develop the list of priorities and baseline data for observation during the EPU power escalation. By comparing the observed pipe vibrations/displacements at various power levels with previously established acceptance criteria, potentially adverse pipe vibrations were identified, evaluated and resolved.

Based on a review of the tubing/support configuration, the layout is such that tubing stress levels remain below the endurance limit while vibrating at these displacement levels.

Engineering has reviewed the vibration and thermal expansion data from each of the applicable plateaus and determined that all lines in the monitoring program have met their acceptance criteria.

Thermography Checks and Temperature Profiles

Temperature monitoring of the Main and Auxiliary Transformers and the Isophase Bus duct using thermography was performed at the 30%, 50%, 75%, 87%, 89%, 92%, 95%, 98% and 100% power levels. This thermography checks were performed to ensure none of the equipment was overheating due to the EPU power increase. The test data was evaluated and found that all temperatures were below design limits for 100% generator output.

Steam Generator Level / Feedwater Flow Dynamic Test

Feedwater Regulating Valve Performance tests were performed at the 30%, 87% and 95% power levels. Each of the SG level control systems was tested to demonstrate a stable control system after EPU modifications to the FW system and level control system. The test imposed a 3-5% level deviation and determined if the automatic SG level control system restored to level program. The test data were evaluated and found that each of the SG level control system met the acceptance criteria.

10% Load Ramp

Two 10% load changes are required at a ramp rate of 1%/min. The first test was a down power equivalent to a 10% turbine load change starting at 100% power +0% - 1%. The second load change was a load increase equivalent to a 10% turbine load change starting at approximately 90%.

As stated above in Section II, the purpose of these tests are to provide additional confidence in the validity of the analytical models and assumptions used in the analysis of plant modifications and integrated plant response to transients, and also verify that no new thermal hydraulic phenomena or adverse system interactions are created by the EPU.

The dynamic behavior of the various plant control systems were observed and evaluated against Level 1 and Level 2 acceptance criteria to ensure that the combination of increased EPU power and changes to the plant configuration (EPU modifications) did not result in an unacceptable aggregate impact. Test acceptance criteria were:

- RCS average temperature, pressurizer pressure, and pressurizer water level will be controlled to the programmed values.
- Steam generator water level will demonstrate good feedwater level control and maintain acceptable margin to the trip level setpoint.
- Steam generator water level should return to programmed level setpoint within ±2 percent narrow range with dampening oscillations within 15 to 20 minutes.
- Nuclear power peak overshoot/undershoot should be less than 3 percent reactor thermal power.

10% Down Power

Unit 4 performed the 10% ramp down power on 7/25/13 from approximately 99.7% rated thermal power (RTP) to approximately 89.0% over a 10 minute period reducing turbine load by 90.2 MWe. This down power achieved a ramp rate greater than 1%/min and reduced power greater than the prescribed 10% turbine load.

A review of system transient data for the 7/25/13 10% load reduction shows that:

- RCS average temperature, pressurizer pressure, and pressurizer water level were controlled to their programmed values.
- Steam generator water level demonstrated good feedwater level control and maintained acceptable margin to the trip level setpoint.
- Steam generator water level returned to programmed level setpoint within ±2 percent narrow range with dampening oscillations within 15 to 20 minutes.
- Nuclear power peak overshoot/undershoot was less than 3 percent reactor thermal power.

In addition, Level 1 and 2 criteria for each of the above control systems were met since each controlled their respective process parameter within technical specification limits with margin. There were no new thermal hydraulic phenomena or adverse system interactions observed. Adequate margin to trip and engineered safety system actuation was maintained throughout the transient.

All acceptance criteria were met for the 10% load down power test.

10% Power Increase

Unit 4 performed the 10% ramp power increase on 7/25/13 from approximately 88.9% RTP to approximately 99.2% over a 10 minute period increasing turbine load by 86.3 MWe. The power increase achieved a ramp rate greater than 1%/min and increased turbine load greater than the prescribed 10% turbine load. Starting at 1.1% below 90% power provided ample margin to avoid over power at the end of the ramp.

The initial condition for the ramp test is specified in Table 2.12-1 above which covers both steady state plateau data gathering done at 3% intervals and the dynamic 10% ramp tests. Since the tolerance covers both types of test and the steady state plateaus are only 3% apart, a 1% tolerance is needed to ensure an adequate power increase that will result in observable changes in parameters.

Table 2.12-1 has "(down)" applied to the box for the 10% ramp test under the 100% column. The intent of this note is to indicate that the 1% tolerance on initial power does not apply to the ramp up from 90% to minimize the potential for an overpower condition to occur due to turbine control system tolerance. As stated in References 1, 2 and 3, the intent of the test is to determine there are no adverse system interactions or new thermal hydraulic phenomena introduced to plant systems from EPU changes and to sufficiently test NSSS and BOP control systems.

Each of the system parameters impacted by the test is affected by a change in either Tref or Tavg. The turbine, control rod, pressurizer level, pressurizer pressure and steam generator level control system will respond to the magnitude and rate of either Tavg or Tref change either through thermodynamic/hydraulic changes. The maximum Tavg change is a function of the turbine ramp rate, reactivity feedback from the core (power defect) and control rod system response to Tref. Given the same ramp rate and control system setting for a ramp test done at two different initial power conditions, only power defect would affect Tavg response. A review of the power defect curves for Unit 3 shows that between 100 and 80% power, the power defect is linear with reactor power. Therefore, the amount of negative reactivity added to the core per degree change of Tavg will be the same anywhere between 100% power and 80%. Therefore, starting the test at a slightly lower initial power will not change the magnitude or rate of RCS temperature change and the challenge to the control systems will be the same. Secondary system response is a function of FW flow which is linear with turbine load/steam flow. There is a slight increase in the differential pressure across the FW regulating valve as FW flow decreases which make it more challenging for S/G level control. Therefore, starting the test at a slightly lower initial power level will not change the magnitude or rate of FW and steam flow changes. Since it has been concluded that the slightly lower initial power level will not reduce the challenge to control systems and thermal hydraulic responses, the power reduction performed on 7/25/13 is acceptable for demonstrating integrated plant response to transients and verifying that no new thermal hydraulic phenomena or adverse system interactions are created.

A review of system transient data for the 7/25/13 10% load increase shows that:

- RCS average temperature, pressurizer pressure, and pressurizer water level were controlled to their programmed values.

- Steam generator water level demonstrated good feedwater level control and maintained acceptable margin to the trip level setpoint.

- Steam generator water level returned to programmed level setpoint within ±2 percent narrow range with dampening oscillations within 15 to 20 minutes.

- Nuclear power peak overshoot/undershoot was less than 3 percent reactor thermal power.

In addition, Level 1 and 2 criteria for each of the above control systems were met since each controlled their respective process parameter within technical specification limits with margin. There were no new thermal hydraulic phenomena or adverse system interactions observed. Adequate margin to trip and engineered safety system actuation was maintained throughout the transient.

All acceptance criteria were met for the 10% load down power test.

Plant Radiation Surveys

The purpose of the survey is to verify plant areas remain as low as reasonable achievable (ALARA) per 10 CFR 20 and doses remain within current limits. 10 CFR 20 criteria is met by confirming doses remain low in normally accessible areas, and dose rate postings within each area are maintained. Turkey Point also has design basis dose limits. Per UFSAR Chapter 11.2 and Chapter 8A normal operation doses are considered in shielding design and environmental qualification of electrical equipment. Normal operational doses limits are based on the radiation zones identified in UFSAR Table 11.2-1. In addition, the secondary shield wall inside containment is designed to attenuate normal operational doses such that the dose rate at the outside of the containment wall is less than 1 mrem/hr.

Radiation surveys were performed at the 87% power plateau for a baseline at the pre-EPU 100% equivalent power level and at the EPU 100% power level in accessible areas that were expected to have increased dose rates. The expected increase in dose rates should be approximately 15% for those areas that are subject to nitrogen-16 gamma or neutron flux.

Inside Containment

The dose rates measured just inside the secondary shield wall (RCS piping area) and outside the secondary shield walls showed measured dose rate increases in some locations that are larger than expected. A review of historic survey maps for the same areas from 2008 prior to EPU show readings higher than at EPU 100% power. The surveys in the primary piping loop area are performed by mounting a detector on the end of a pole and inserting the pole into the loop area while the RP technician stands behind the secondary shield wall. Using this type of measurement cannot reproduce the same survey dose point with any precision. A review of the dose rates inside the secondary shield wall shows significant differences based on minor changes in physical locations. Measurements outside the secondary shield wall have similar challenges due to the streaming affect from shield wall penetrations. A small change from target survey locations will change dose significantly and the dose rates at 100% power do not allow for a precision map to be made.

Containment is classified as a locked high radiation area. The measured dose rates did not increase the level of the posting beyond a locked high radiation area. The controls imposed by the Technical Specifications and 10 CFR 20 for locked high radiation areas will maintain personnel dose ALARA.

Inside the secondary shield wall (RCS loop area) the EQ program assumed a dose rate of 60 R/hr and the measured value at the entrance to the RCS piping area was 0.5 R/hr. The EQ program assumed a dose rate of 1.125 R/hr outside the secondary shield wall and the maximum measured dose was 0.4 R/hr.

The dose rates measured adjacent to the containment wall did not change from measured values prior to EPU and remain less than 1 mrem/hr.

Outside Containment

Measurements outside containment did not show any appreciable increase in dose except for the Charging pump room and pipe and valve room. The charging pump room radiation measurements showed that half of the locations had decreased radiation levels and half increased in radiation level. The largest increase went from 1.0 mrem to 5 mrem near the Charging pump. This was the highest dose for the locations that increased was found near the letdown line which went form 48 to 66 mrem/hr. In the pipe and valve room the largest increase went from 26 mrem/hr to 34 mrem/hr near the RHR penetration.

While these areas exceeded the expected 15% increase, the actual dose rates are very low and do not affect the postings or radiation zone classifications for any area. The Auxiliary Building areas surveyed are controlled radiation areas. The measured dose rates do not increase the level of the posting beyond a radiation area. The controls imposed by the technical specification and 10 CFR 20 for radiation areas will maintain personnel dose ALARA. The EQ program assumed dose rates at the maximum allowed for the radiation zone classification. All surveyed areas remained below the radiation zone classifications.

The measured dose rates inside and outside containment at 100% EPU power meet 10 CFR 20 and Turkey Point limits.

Plant Temperature Surveys

Plant ambient temperature survey data was collected with data recorders in those areas potentially impacted by EPU equipment heat loads. Most areas did not see any appreciable temperature change. All area temperatures remain below the general outside area design limit of 104°F except the condensate pump area and main feedwater pump room.

As part of the EPU design changes for the Condensate pumps general area supply and exhaust fans were added for the condensate pump area. However, temperatures were observed between 120°F and 135°F even with the addition of the fans. The condensate pump and motor design temperature limits are above an ambient temperature of 140°F except for the motor lower radial bearing. The lower radial bearing does not have an oil cooler and relies on ambient cooling. The bearing has experienced alarms which actuate at a bearing temperature of 185°F. A review of the air flow in the condensate pump area shows some portions may not be receiving adequate flow. Temporary fans have been installed to augment air flow to the low flow areas and it is maintaining bearing temperatures below their alarm point until a permanent solution can be implemented. Procedural guidance directs shutting down the pump if bearing temperatures reaches 200°F. The high area temperature condition has been entered into the Turkey Point Corrective Action Program for resolution.

The Steam Generator Feed Pump (SGFP) Room temperature reached 114°F during some periods. While the EPU did increase FW flow the heat load to the FW pump room did not increase. This higher room temperature condition is a pre-EPU issue which has been compensated for by vane axial fans which provide forced circulation through the SGFP Motors. An exhaust plenum is located above each SGFP motor and is attached to 25,000 cfm exhaust fans. The additional fans maintain the FW pump motors below their design maximum operating temperatures. All other equipment within the FW pump room is rated for temperature well above 114°F. Based on past plant operation, the elevated temperatures have not caused a reduction in FW pump reliability.

Leading Edge Flowmeter (LEFM) Commissioning

As described in Reference 1, the Turkey Point EPU project included a 1.7% Measurement Uncertainty Recapture (MUR) thermal power increase. To achieve the MUR power increase of 1.7%, the Cameron LEFM CheckPlus™ ultrasonic flow measurement instrumentation was installed to improve feedwater flow measurement accuracy. An individual LEFM CheckPlus™ system flow element (spool piece) was installed in each of the three main feedwater lines and was calibrated in a site-specific model test at Alden Research Laboratories with traceability to National Standards.

The LEFM CheckPlus™ system was installed and commissioned in accordance with FPL procedures and Cameron installation and test requirements. LEFM CheckPlus™ commissioning included verification of ultrasonic signal quality and evaluated the actual plant hydraulic velocity profiles as compared to those documented during the Alden Research Laboratories testing. Final verification of the site-specific uncertainty analyses occurred as part of the LEFM CheckPlus™ system commissioning process. The commissioning process provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation which satisfies licensing commitment 13 in Section 3.3 (a) of the EPU SER. In addition to the Cameron commissioning test and evaluations, FPL evaluated LEFM performance as follows:

A review of feedwater parameters for LEFM and the Venturi based measurements were completed to determine deviations and reasonableness of the Venturi correction factor. There are no operational alarms or other deficiencies noted. Steam Generator Heat rates and calorimetric calculations were performed using the LEFM and Venturi based data. The LEFM ultrasonic flow transmitters are performing as expected. The LEFM is showing a 0.52% greater thermal power than the Venturi at 100% power. The Venturi LEFM correction factor adjusts the Venturi flowrate down such that the 0.52% deviation is eliminated and Venturi flow matches LEFM flow.

V. Summary

The test data collected during EPU startup and power ascension and summarized in this report demonstrates that all major SSCs performed as predicted or has been found acceptable for operation at EPU conditions. The final testing results have shown there was no adverse impact to the performance of the unit. The final 100% EPU startup and power ascension test data satisfied all acceptance criteria except for HDT discharge valve position which was found acceptable after evaluation. Copies of the completed EPU startup and power ascension test procedures are available on site for review.

VI. References

1. J. Page (NRC) to M. Nazar (FPL), Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Extended Power Uprate (TAC Nos. ME4907 and ME4908), Accession No. ML11293A365, June 15, 2012.
2. M. Kiley (FPL) to J. Page (NRC) (L-2010-113), License Amendment Request No. 205: Extended Power Uprate (EPU), (TAC Nos. ME4907 and ME4908), Accession No. ML103560169, October 21, 2010.
3. M. Kiley (FPL) to J. Page (NRC) FPL (L-2011-101), Turkey Point Units 3 and 4 Docket Nos. 50-250 and 50-251 Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Balance of Plant Issues, Accession No. ML11105A146, April 14, 2011.
4. M. Kiley (FPL) to U. S. Nuclear Regulatory Commission (NRC) (L-2013-218), Renewed Facility Operating License No. DPR-41 Extended Power Uprate Cycle 27 Startup Report, July 16, 2013.

Table 1
Primary and BOP Major Parameter Summary

Primary Major Parameter	**89%**	**92%**	**95%**	**98%**	**100%(1)**
Avg Power %	89.2	92.3	94.7	97.2	99.2
Avg Thot °F	605.0	607.2	609.3	611.0	612.4
AvgTcold °F	546.7	547.1	547.4	547.5	547.4
Avg Tavg °F	575.9	577.1	578.4	579.2	579.9
SG A Press psig	800	797	793	788	783
SG B Press psig	800	797	794	789	785
SG C Press psig	799	796	792	788	783
Pzr Avg % Level	52.5	52.4	55.0	55.8	56.3
Turbine Inlet Avg Press (TIP) psig	579.2	600.6	624.1	647.2	662.0
Tref °F	575.3	576.3	577.6	578.6	579.2
Containment Temp °F Highest	108.1	109.5	108.6	105.3	110.0
BOP Major Parameter					
SG A Level %	47.3	47.1	47.2	47.1	47.0
SG B Level %	48.3	48.1	47.8	47.9	47.9
SG C Level %	48.4	48.5	48.4	48.4	48.3
Avg Steam flow MPPH	3.36	3.48	3.62	3.74	3.84
Avg FW flow MPPH	3.44	3.57	3.69	3.79	3.89
Final Venturi FW Temp °F	430.5	433.1	436.2	439.1	440.8
Condenser Backpressure in-Hg	2.64	2.84	2.70	2.50	3.10
Generator Output Mwe	757.3	782.6	817.1	854.7	855.3
Thermal Output MWt	2346.4	2420.7	2501.1	2569.9	2640.0

(1) All feedwater flow and thermal output values are based on the Venturi flow meter