Entergy

Entergy Nuclear South
Entergy Operations, Inc.
17265 River Road
Killona, LA 70057-3093
Tel 504-739-6715
Fax 504-739-6698
rmurill@entergy.com

Robert J. Murillo
Licensing Manager, Acting
Waterford 3

W3F1-2005-0083

December 8, 2005

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

Subject: Supplemental Startup and Power Report for Cycle 14
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

REFERENCES:
1. NRC letter to Entergy dated May 9, 2005, "Waterford Steam Electric Station, Unit 3 – Issuance of Amendment Re: Modification of Technical Specifications (TS) 5.3.1, Fuel Assemblies, TS 5.6.1, Criticality, TS 6.9.1.11.1, Core Operating Limits Reports, and Deletion of TS Index (TAC No. MC3584)"
2. NRC letter to Entergy dated April 15, 2005, "Waterford Steam Electric Station, Unit 3 – Issuance of Amendment Re: Extended Power Uprate (TAC No. MC1355)"

Dear Sir or Madam:

In letter W3F1-2005-0067, dated September 9, 2005, Entergy Operations, Inc. (Entergy) provided a summary report for plant startup and power escalation testing for Waterford 3 conducted at the beginning of Cycle 14. As stated in the letter, a detailed report could not be provided due to the impact of hurricane Katrina on plant resources. Also, as stated in the letter, a more detailed report of the Extended Power Uprate test results, including a discussion on the test maneuvering test results, would be provided in a Supplemental Startup Report pursuant to Waterford 3 Technical Specification 6.9.1.3.

The Supplemental Startup Report is provided in Attachment 1 which satifies the requirements of Technical Specification 6.9.1.3.

There are no new commitments contained in this submittal.

If you have any questions or require additional information, please contact Greg Scott at 504-739-6703.

Sincerely,

RJM/GCS/cbh

Attachment

IE26

cc: Dr. Bruce S. Mallett
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Waterford Steam Electric Station Unit 3
P.O. Box 822
Killona, LA 70066-0751

U. S. Nuclear Regulatory Commission
Attn: Mr. N. Kalyanam
Mail Stop O-07D1
Washington, DC 20555-0001

Wise, Carter, Child & Caraway
ATTN: J. Smith
P.O. Box 651
Jackson, MS 39205

Winston & Strawn
ATTN: N.S. Reynolds
1700 K Street, NW
Washington, DC 20006-3817

Morgan, Lewis & Bockius LLP
ATTN: T.C. Poindexter
1111 Pennsylvania Avenue, NW
Washington, DC 20004

Attachment 1

W3F1-2005-0083

Supplemental Startup and Power Report for Cycle 1

Startup and Power Escalation Report for Cycle 14

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Startup and Power Escalation Report for Cycle 14

1.0 INTRODUCTION

1.1 Summary

In accordance with the requirements of Waterford 3 Technical Specification 6.9.1.a, Entergy submitted the Startup and Power Escalation Report for Cycle 14 Waterford Steam Electric Station, Unit 3 (Ref 5.6) on September 9, 2005. The report contained a general summary of the WSES-3 Cycle 14 startup test program and test results as they pertained to a core thermal power uprate from 3441 to 3716 MWt.

Testing specified in W3F1-2004-0004, Supplemental Information, Extended Power Uprate – Startup Testing, (Ref 5.5) was addressed. Special test procedures were implemented in combination with existing plant procedures, as described in this report. Plant surveillance test procedures were used, to the extent possible, to satisfy required testing. The test program included pre-critical tests as well as those conducted during low power physics testing (LPPT), power ascension and at full power. While all tests performed as a part of this program were completed satisfactorily, not all test results were included in this summary. Only the tests deemed necessary to demonstrate performance at the uprated condition and conformity with the planned testing specified in W3F1-2004-0004, Supplemental Information, Extended Power Uprate – Startup Testing, were included.

Due to hurricane Katrina and the redirection of Entergy personnel to plant recovery activities following the hurricane, a more detailed summary report for the EPU test results was not complete at the time of the initial report. A brief summary of the EPU test results, along with a detailed report on the core physics testing, was submitted at that time. In addition, the plant maneuvering test had not yet been completed at the time of the submittal.

This report provides a detailed report of the EPU test results, and a discussion of the analysis of plant transients used to satisfy the objectives of the plant maneuvering test. Since a detailed discussion of core physics testing was provided in the initial Startup Test Report, no further discussion of core physics testing is provided.

The completed Extended Power Uprate (EPU) Test program has demonstrated that the analyses, modifications, and adjustments necessary for EPU have been properly performed. Additional data has been collected to benchmark the analysis against the actual integrated performance of the plant. The results of testing performed during Power Ascension have been reviewed by the Onsite Safety Review Committee.

The results of the testing and equipment performance data gathering have demonstrated acceptable continued plant operation at the uprate power level of 3716 MWt.

1.2 Acceptance Criteria

For each test performed in the power ascension test program, test conditions and associated acceptance criteria were defined within the test. For tests utilizing existing plant procedures and surveillances, acceptance criteria were updated for uprated conditions. For special tests developed for power uprate start-up testing, two levels of acceptance criteria were developed. Level 1 criterion was associated with safe unit operation, and required a halt to power ascension. Level 2 criteria were associated with system/component performance expectations, and were evaluated prior to continuing to the next power plateau.

2.0 POWER ASCENSION

2.1 NSSS Plant Data Record

PURPOSE:

The purpose of this test was to provide a permanent baseline data record of plant parameter indications from zero power to full power operation (3716 MWt) during steady operations.

METHOD:

Data collection commenced with the plant at hot zero power. The large data set was collected automatically from the Plant Monitoring Computer, and stored in an Excel spreadsheet. A small subset of the data was collected manually. A data set was collected at every 10% reactor power (i.e. 10%, 20%, 30%, etc.) from hot zero power to full power operation. Additional data sets were also collected at 92.59% (3441 MWt, the pre-EPU core rated thermal power), 95%, 97.5%, and 100% rated thermal power. In part, the data set consisted of:

- Reactor power and operating limits data
- RCS (pressure, temperature, boron, etc.) data
- RCS differential pressure data
- Core exit thermocouple and heated junction thermocouple data
- Secondary plant data

- Incore instrumentation data
- CEA data
- Core Protection Calculator (CPC) data
- Turbine-generator data
- Plant chemistry data (above 92.59%)

Data collection was terminated upon completion of power ascension testing.

RESULTS:

The required data was gathered at the specified intervals. The data that was collected provides a plant baseline record for future reference.

Prior to placing the Main Turbine in service, the Steam Bypass Control System (SBCS) was controlling RCS temperature high in the desired band. This condition was evaluated to be acceptable for continued plant operations (Ref 5.7). Subsequent investigations discovered that one input to the SBCS, Main Steam Header Pressure, was out of calibration. This condition has been corrected, and the SBCS was verified to control RCS temperature acceptably during the post Hurricane Katrina plant startup.

CONCLUSION:

A substantial data base of significant plant parameters was established for plant conditions corresponding to reactor power ranging from hot zero power to 100%. This data will be retrievable for future reference, and has been provided to Training for comparison with simulator performance. All test objectives and acceptance criteria were satisfactorily met.

2.2 Transient Data Record

PURPOSE:

The transient data record provided a plant baseline data record during the slow initial power increases of the Waterford 3 plant. The data provides an overview of primary and secondary plant loads and operating conditions and how they changed during power increases.

METHOD:

Data collection commenced with the plant at hot zero power. The large data set was collected automatically from the Plant Monitoring Computer, and stored in an Excel spreadsheet. A data set was collected every 5 minutes from hot zero power to full power operation. In part, the data set consisted of:

- Plant Power
- RCS Temperature
- Reactor Power Distribution
- Operating Margin
- CEA Positions
- Turbine Load
- SBCS Steam Loads
- Steam Generator Energies

Data collection was terminated upon completion of power ascension testing.

RESULTS:

The required data was gathered at the specified intervals. The data that was collected provides a plant baseline record for future reference.

CONCLUSION:

Representative baseline data was collected during all power increases from zero to 100% full power. This data will be retrievable for future reference, and has been provided to Training for comparison with simulator performance. All test objectives and acceptance criteria were satisfactorily met.

2.3 Nuclear and Thermal Power Calibrations

PURPOSE:

The objective of the nuclear and thermal power calibration was to calibrate Excore linear power, CPC thermal power, and CPC nuclear power to a standard measurement of core power.

METHOD:

Initial conditions were established with the reactor at steady-state conditions. A standard measurement of core power was then determined by one of the following methods:

A. Up to approximately 35% rated thermal power, reactor power was calculated from a primary system calorimetric measurement (i.e., RCS delta-T power measurement).

B. Above 35% rated thermal power, reactor power was obtained from a secondary energy balance calculation performed by the Core Operating Limit Supervisory System (COLSS).

Utilizing the standard power, a new voltage output from the Excore linear amplifier was determined. The amplifier gain was then adjusted as necessary per existing plant procedures to obtain the new voltage.

Calibration of CPC nuclear power (PHICAL) and thermal power (BDT) was accomplished by changing the respective values of the associated CPC addressable constant. These constants were computed and changed as necessary using existing plant procedures.

RESULTS:

The nuclear and thermal power calibration procedure was performed at the following power levels during power ascension: 20%, 50%, 80%, 92.59%, 95%, 97.5%, and 100%. In each case, the PPS Excore linear power and CPC powers were verified to be within the required tolerance, or were calibrated to within the required tolerance.

CONCLUSION:

The PPS and CPC power indications were successfully calibrated to the standard indication of reactor power.

2.4 Initial Turbine Startup

PURPOSE:

The purpose of this test was to verify proper operation of the turbine and generator by accelerating the turbine to operating speed, synchronizing the unit, and loading the unit to 100% of rated load. In addition, various turbine protective devices were tested for proper operation. Finally, a baseline record of turbine operation was established as part of the BOP Data Record and the Transient Data Record.

METHOD:

Testing and operation of the main turbine was performed in the same manner as the initial turbine startup following every refueling outage. The turbine was latched, and accelerated to 520 RPM, where a turbine walkdown was performed. The turbine was then accelerated to 1800 rpm and a turbine trip test was performed to verify the operation of the turbine valves. An overspeed trip test of the turbine was then performed by raising turbine speed to the overspeed trip setpoint and observing a turbine trip.

The turbine was then latched and accelerated to 1800 rpm, at which time the main generator was synchronized to the grid and loaded to 100% of rated load per plant procedures.

RESULTS:

All acceptance criteria for initial turbine startup were met.

CONCLUSION:

The turbine and generator operated as expected. The turbine protective devices performed as designed. A baseline data record of turbine loading has been created as part of the BOP Data Record and the Transient Data Record. All acceptance criteria were met.

2.5 Linear Power Subchannel Calibration

PURPOSE:

Based on previous operating experience and the core reload design process, the linear power subchannel gains will be adjusted prior to startup. The methodologies used in previous cycles will remain unaffected.

METHOD:

Excore linear power subchannel amplifier gains were adjusted prior to initial criticality using Startup Test Predictions (Ref 5.8). The Excore detector flux ratio (BOC 14/BOC 13) was 1.2800. Adjustments are made to insure that all 3 subchannels are adjusted equally, maintaining the validity of the Cycle Independent Shape Annealing Matrix (CISAM) from Cycle 13. The subchannel gains and the CISAM are then validated for Cycle 14 during performance of NE-002-100, Fast Power Ascension Data Collection and Analysis, between 15% and 70% RTP.

RESULTS:

Excore linear power subchannel amplifier gains were satisfactorily adjusted prior to initial criticality using Startup Test Predictions. The amplifier gains were validated during performance of NE-002-100, Fast Power Ascension Data Collection and Analysis.

CONCLUSION:

All test objectives and acceptance criteria were met. No additional adjustments of subchannel amplifier gains were required beyond the pre-criticality adjustments specified by the Startup Test Predictions.

2.6 Process Variable Intercomparison

PURPOSE:

The purpose of this test was to demonstrate that the inputs and appropriate outputs of the Plant Protection System (PPS), the Core Protection Calculators (CPCs), and the Plant Monitoring Computer (PMC) were in satisfactory agreement with one another.

METHOD:

Plant conditions were stabilized at four test plateaus – 92.59%, 95%, 97.5%, and 100% RTP. Data from each of the four sources (PPS, CPCs, PMC, and permanent plant instrumentation) were simultaneously gathered for each of the following parameters:

1. RCS cold leg temperature
2. RCS hot leg temperature
3. RCP differential pressure
4. RCP speed
5. RCS pressure
6. Pressurizer level
7. Steam generator level
8. Steam generator pressure
9. Steam generator primary side differential pressure
10. Reactor vessel differential pressure
11. Containment pressure
12. Refueling water storage pool (RWSP) level

Based upon the data gathered for each parameter, a target value was calculated as the average of the readings from the most reliable source, the CPCs. The deviation of each recorded value from this target value was calculated and compared to the specified tolerance to determine acceptability.

RESULTS:

All safety-related indications met comparison limitations consistent with current analysis, design requirements, and past (pre-EPU) performance.

One instrument, RC ITE0115, was out of service at the time of the test due to failing a post modification channel check (i.e. process variable Intercomparison). This instrument supplies a redundant indication of RCS Cold Leg 1B temperature to COLSS. The point was removed from scan in the COLSS program at the time of the test. After repair and post maintenance testing, this instrument will be returned to service in accordance with routine maintenance procedures (Ref 5.9).

Expected divergence was observed in the indications of RCS Hot Leg temperature (+3.4/-5.0 of average value). This was anticipated due to hot leg flow streaming, which is a well known industry phenomena. Hot leg flow streaming exists at Waterford 3 as it does in many other nuclear power plants. Waterford 3 utilizes the DELSTRAT code to account for the effects of flow streaming in the performance of RCS Flow Rate Technical Specifications. A review of previous cycle data indicates that the divergence in RCS Hot Leg temperature is consistent with that observed at the beginning of cycle for previous core reloads.

CONCLUSION:

All safety-related indications exhibited satisfactory agreement with one another. The process variable accuracies support current analysis, and are consistent with both design requirements and with past (pre-EPU) performance.

2.7 Radiation Surveys

PURPOSE:

This test obtained selective dose rate surveys inside the Reactor Auxiliary Building (RAB) at 92.59%, 95%, 97.5% and 100% RTP, and inside the Reactor Containment Building at 100% RTP. The purpose of this test was to identify any changes in radiological conditions outside of the containment at the uprated conditions, and to establish the adequacy of the biological shield inside the containment.

METHOD:

Portable neutron and gamma survey equipment was used in performing all phases of the biological shield survey.

RESULTS:

Radiation levels in the Reactor Auxiliary Building and the Reactor Containment Building were within the estimates specified in the FSAR. No change in radiological postings was required due to extended power uprate.

CONCLUSION:

Extended Power Uprate has resulted in minimal impact on radiological conditions. The test demonstrated that the biological shield is acceptable at the uprated power level. All radiation levels were within those specified in the FSAR.

2.8 CPC Process Noise

PURPOSE:

The purpose of this test was to evaluate the impact of power uprate on the process noise in the CPC system at 100% RTP.

METHOD:

A collect log was set up to sample all CPC inputs, and DNBR and LPD outputs, on all 4 CPC channels. Data was collected every 15 second for 30 minutes from steady state conditions. The percent oscillation was determined by dividing the difference between the maximum and the minimum value by the average value for each parameter. The percent oscillation for each parameter was determined at the uprated licensed power condition (3716 MWth), and at the pre-uprate licensed power condition (3441 MWth).

The percent oscillation for each parameter at the uprated licensed power condition (3716) was then compared to the percent oscillation for each parameter at the pre-uprate licensed power condition (3441 MWth) to determine the impact of power uprate on process noise.

RESULTS:

The below chart displays the maximum input and output oscillation observed on any CPC (pre-EPU and post EPU conditions), the maximum change in input and output oscillation from the pre-EPU to the post EPU condition observed on any CPC, and the average change in input and output oscillation from the pre-EPU to the post EPU condition observed on all CPCs.

INPUTS	Pre-EPU Condition	Post EPU Condition	Change	Post EPU Parameter
Max Input Oscillation	0.81%	0.85%		CPC D Upper Excore Det.
Max Change in Input Oscillation			0.17%	CPC D Upper Excore Det.
Ave change in Input Oscillation			-0.06%	

OUTPUTS	Pre-EPU Condition	Post EPU Condition	Change	Post EPU Parameter
Max Output Oscillation	4.14%	3.71%		CPC C DNBR
Max Change in Output Oscillation			0.29%	CPC C DNBR
Ave Change in Output Oscillation			-0.09%	

CONCLUSION

The results show that EPU has a negligible affect on CPC process noise. For parameters that exhibited an increase in oscillation, this increase was a small fraction of the measured value. EPU has a negligible affect on CPC calculation of LPD and DNBR.

2.9 COLSS Power / Flow Verification Data Record

PURPOSE:

This test collected data between 20% and 100% RTP to be used in determining the proper constants for the COLSS calibrated turbine power calculation.

METHOD:

Data was taken at 30 second intervals from 20% to 100% RTP. Data was discarded where cold leg temperature was more than +/- 2 °F from program. A least squares fit of COLSS turbine first stage pressure (TFSP) verses the best COLSS power indication was calculated to determine the coefficients (G1 through G4) of a third order polynomial defining the relationship of TFSP to core thermal power. The best COLSS indication of thermal power was determined based upon the best calculated accuracy of indicated power at each power level as follows:

0-35%	BDELT (Delta-T power)
35% - 40%	FWBSRAW (Secondary calorimetric based on feedwater flow rate as measured by the venturi flow meter)
40% - 100%	USBSRAW (Secondary calorimetric based on feedwater flow rate as measured by the Ultrasonic flow meter)

RESULTS:

Calibrated turbine power polynomial coefficients G1 through G4 were determined and entered into COLSS. Final indication of Turbine First Stage Pressure power (BTFSP) met the following acceptance criteria:

20% - 40%	BTFSP within 3% of actual power
40% - 80%	BTFSP within 2% of actual power
80% - 100%	BTFSP within 1% of actual power

CONCLUSION:

All test objectives and acceptance criteria were met. Values for G1 through G4, calibrated turbine power polynomial coefficients, were properly entered into COLSS

2.10 COLSS Secondary Pressure Loss Term

PURPOSE:

The purpose of this test was to validate, for power uprate conditions, the COLSS algorithms for which calculate feedwater pressure. These algorithms are required due to the unavailability of feedwater pressure indication on the plant computer system downstream of the feedwater flow control valves. The algorithms model the pressure drop between the feedwater inlet to the steam generator as a function of

system flow (specifically, steam flow) in steady state conditions. COLSS then uses live steam header pressure and steam flow data to calculate the feedwater pressures for use in the calculations of secondary calorimetric power.

This test collected live data for feedwater pressure using temporarily installed gauges, and compared that to the COLSS calculated feedwater pressure to validate the accuracy of the COLSS algorithms.

It should be noted that the original algorithms for COLSS calculated steam pressure have been replaced with a live input from steam generator pressure instruments.

METHOD:

Values for feedwater pressure (live data) and COLSS calculated feedwater pressure (FWP) were recorded with the plant at steady state conditions at 92.59% (pre-EPU licensed power) and 100% RTP. Temporary pressure gauges were installed as close as practicable to the feedwater inlet of the steam generators. These gauges were used to collect the feedwater pressure (live data). Data was averaged over a 10 minute period.

RESULTS:

Acceptance criteria established for this test was that COLSS calculate feedwater pressure (FWP) was within 50 psi of measured feedwater pressure (live data). The results were as follows:

	92.59%		100%	
	FW Train A	FW Train B	FW Train A	FW Train B
FWP (COLSS Calculated)	839.7	856.6	849.3	866.1
Live Data (gauges)	855.9	866.8	867.8	878.9
Difference	16.2	10.2	18.5	12.8

CONCLUSION:

COLSS calculated feedwater pressures were compared to measured values, and found to be within the specified tolerances at 92.59% (pre-EPU licensed power) and 100% RTP. No modifications of the COLSS algorithms were necessary. The test objectives were satisfactorily met.

2.11 BOP Data Record

PURPOSE:

The purpose of this test was to collect data relative to secondary plant systems and components to establish a baseline for future performance comparisons and analysis.

METHOD:

Data collection commenced with the plant at hot zero power. The large data set was collected automatically from the Plant Monitoring Computer, and stored in an Excel spreadsheet. A small subset of the data was collected manually. Data was collected at steady state conditions and during the slow power ascension (less than 3%/hr) from hot zero power to full power operation. In part, the data set consisted of parameters from the following:

- Extraction Steam
- Main Steam
- Primary System Parameters
- Steam Generator Feedwater Pumps
- Gland Sealing Steam
- Feedwater Heater Drains
- Main Condenser
- Main Turbine/Generator
- Steam Generators
- Condensate System
- Feedwater System

RESULTS:

A data set was collected at every 10% reactor power (i.e. 10%, 20%, 30%, etc.) during power ascension from hot zero power to full power operation. This data was collected during a slow power ascension (less than 3%/hr), rather than at steady state condition. Additional data sets were collected at steady state conditions at 68%, 92.59% (3441 MWt, the pre-EPU core rated thermal power), 95%, and 97.5% rated thermal power.

Prior to power ascension, Siemens Westinghouse reported an error in the design of the newly installed high pressure turbine that would result in a lower admission pressure. The most notable affect of that error was a final turbine governor valve position of approximately 50% open, vice the expected 80% open at 100%. This

condition has been entered into the plant corrective action process for long term corrective action (Ref. 5.10).

Shell Drain Tank (SDT) 2B level exceeded the acceptance criteria at 100% RTP. The level in the SDTs is a function of system parameters (flow, etc.) This condition was evaluated, and determined to be acceptable for continued operation at the EPU conditions. The level remains on scale, and sufficient margin exists to High Level Alarm. This condition has been entered into the plant corrective action program (Ref. 5.11).

The Main Turbine Governor Valves were observed to be oscillating in response to grid frequency oscillations on two occasions above 92.5% RTP. Investigation revealed that a previous change to a Digital-electro-hydraulic (DEH) program setpoint had not been included in the new DEH program for EPU. This condition was corrected prior to achieving 100% RTP (Ref 5.12).

CONCLUSION:

A substantial quantity of data relative to the secondary plant was collected during the slow power ascension. This data will be retrievable for future reference, and has been provided to Training for comparison with simulator performance. All test objectives and acceptance criteria were satisfactorily met.

Three deficiencies were identified during the performance of this test, and have been discussed above. These conditions have either been corrected, or have been evaluated for acceptability for continued operation at EPU conditions and entered into the plant corrective action program.

2.12 Level 2 Piping Vibration Testing and Non-intrusive Monitoring

PURPOSE:

The purpose of this test was to verify by measurement and/or observation that vibration amplitudes were acceptable for piping systems that may experience changed thermal-hydraulic conditions due to EPU.

METHOD:

All Main Feedwater and Main Steam piping inside containment were monitored utilizing temporarily installed accelerometers. All Main Feedwater and Main Steam piping outside of containment were monitored using hand-held vibration monitors. In total, vibration data was collected from 154 piping locations in the Main Steam and Main Feedwater systems.

The Main Turbine, Main Feedwater Pumps, and Reactor Coolant Pumps were monitored using permanently installed vibration monitoring equipment. Rotating equipment in the steam cycle and in support systems were monitored using hand-held vibration monitors.

Criteria were established for collection and evaluation of the vibration data at each of the four power plateaus from 3441 MWth to 3716 MWth. Acceptance criteria were established based on governing piping codes and standards. Post EPU 100% power (3716 MWth) data was compared to the complete set of baseline data collected just prior to Refueling 13 (March, 2005) at the pre-EPU 100% power (3441 MWth).

An acoustic survey of the Main Condenser was performed at 100% power. The secondary plant was also visually inspected by Operations, Engineering, and Performance Monitoring prior to and following Power Uprate.

RESULTS:

All inspections and data were acceptable. However, five areas of elevated vibration were observed. The data from each location was evaluated by Engineering to be within acceptable limits. Three of these were entered into the site Corrective Action program (Ref 5.13).

One case did involve elevated vibration levels on safety related piping. The evaluation for that piping (Feedwater line inside containment) indicated that a 95% margin exists to the vibration limit for that location. No additional corrective action is required for this condition.

CONCLUSION:

The vibration testing and non-intrusive monitoring demonstrated that vibration levels within the affected piping systems remained within acceptable levels post power uprate.

3.0 POST STARTUP

3.1 Thermal Performance Test

PURPOSE:

The purpose of this test was to collect data and perform calculations to permit comparison of the replacement High Pressure Turbine performance against contract specifications and design criteria.

METHOD:

A pre-EPU and post EPU thermal performance test was conducted at 100% RTP using the methodology of ASME PTC6-1996 (alternative). The results of those tests were corrected to a standard set of conditions, and the electrical outputs compared to determine the increase in electrical power due to EPU and associated plant modifications.

RESULTS:

Prior to power ascension, Siemens Westinghouse reported an error in the design of the newly installed high pressure turbine that would result in a lower admission pressure. The most notable affect of that error was a final turbine governor valve position of approximately 50% open, vice the expected 80% open at 100%. However, Siemens Westinghouse has reported that preliminary results from the performance testing showed an increase in electrical output exceeding the goal of 68 MWe. No other plant deficiencies were identified.

CONCLUSION:

Preliminary results from the thermal performance testing demonstrated an expected increased in electrical output from EPU.

3.2 Steam Generator Moisture Carryover Test

PURPOSE:

The purpose of this test was to collect data to determine the amount of moisture being carried over into the main steam lines from the steam generator. This information was used in calculating the Pre- and Post EPU plant thermal efficiency, and in validating associated constants utilized in the COLSS calculation of secondary calorimetric power.

METHOD:

A pre-EPU and post EPU moisture carryover test was conducted at 100% RTP. The tests were conducted by isolating steam generator blowdown and condensate

makeup, injecting lithium into the feedwater system, and subsequently comparing the concentration of lithium in the steam generators to that in the steam system.

RESULTS:

The result of the tests indicates that moisture carryover increased from 0.211% pre-EPU, to 0.450% post EPU, which exceeds the design carryover of 0.20%. This also exceeded the assumed moisture carryover utilized in COLSS (0.40%), which results in a conservative calculated value of secondary calorimetric power. This condition has been entered into the Waterford 3 corrective action program (ref. 5.16).

The increased moisture carryover is likely caused by the higher steam flow from EPU, and may be aggravated by deposit fouling of the steam generator chevron separators. This condition has been entered into the plant corrective action process to determine the potential long term effects of the measured moisture carryover on secondary plant systems and components.

CONCLUSION:

The results of the tests showed that the moisture carryover has increased slightly coincident with EPU. The current level of moisture carryover is consistent with that used in FAC analysis, and results in a conservative result from the COLSS calculated secondary calorimetric power. This condition is acceptable for use as is. However, this condition has been entered into the plant corrective action process to determine the potential long term effects of the measured moisture carryover on secondary plant systems and components.

3.3 Plant Shutdown and Startup

BACKGROUND:

On August 28, 2005, in anticipation of hurricane conditions on site within 12 hours, Waterford 3 commended a plant shutdown in accordance with site procedures.

Subsequent plant startup and power ascension was performed on September 13, 2005.

METHOD:

A normal plant shutdown was commenced at 1059 on August 28, 2005. Turbine load was reduced at 10 MW/min (approximately 0.8%/min RTP), and the reactor trip breakers were opened at 1316.

A subsequent normal plant startup and power ascension to 100% RTP was commenced at 0328 on September 13, 2005.

RESULTS:

All systems responded as expected during the plant shutdown. Control systems operated satisfactorily in automatic throughout their control range. No new plant deficiencies were identified during the plant shutdown.

Proper operations of the FWCS in both modulate and RTO (reactor trip override) mode was observed. Proper operation of the SBCS in Quick Open/Quick Open Block mode, and modulate mode was observed. Control Element Assemblies were operated in manual for Axial Shape Index (ASI) control. Pressurizer Level Control was operated in automatic during the downpower and post trip. Chemical Volume Control and Main Turbine controls were operated in manual to affect the plant shutdown.

The average rate of the downpower was 0.73%/min. The average rate of the power ascension from 20% RTP to 50% RTP was 0.4%/min. However, due to fuel precondition guidelines, power increase above 50% RTP was limited to 0.33%/min. In comparison, the planned Load Changes Test would have performed only a 10% downpower, followed by a 5% power ascension, at a rate of less than 0.5%/min.

CONCLUSION:

The plant shutdown and subsequent startup resulted in a more extensive transient than that planned for the Load Changes Test; the rate of power change was consistent with that planned for the Load Changes Test; and the Plant Computer System and the Performance Indication data systems automatically stored the data that would have been collected by the Load Changes Test.

Proper response of control systems was verified during the plant shutdown from 100% to 0% RTP (plant trip), and subsequent plant startup and power ascension to 100% RTP.

3.4 Loss of Circulating Water Manual Plant Trip

BACKGROUND:

On 11 November, at 2034, a manual trip of the reactor was performed due to a total loss of Circulating Water (CW) (Ref 5.14). The loss of CW was caused by the failure of a relay in the Circulating Water Pumps control circuitry. After completion of corrective actions, a plant startup and power ascension to 100% RTP was commenced at 0315 on November 14.

PURPOSE:

A review of systems response to this event was performed to demonstrate that the integrated plant control systems operate satisfactorily in automatic to maintain plant parameters within specified limits in response to plant transients.

METHOD:

A manual trip of the reactor was performed due to a total loss of Circulating Water (CW). Main Feedwater Pumps tripped due to low condenser vacuum, and Emergency Feedwater System (EFW) automatically actuated on low Steam Generator Level. Subsequently, the Steam Bypass Control System (SBCS) became unavailable due to the loss of condenser vacuum, and the Atmospheric Dump Valves automatically actuated and modulated to control RCS temperature.

RESULTS:

A review of plant systems response and parameter trends indicate that the integrated plant control systems operated satisfactorily in automatic to stabilize the plant post trip. All safety systems responded as expected, with the following exception.

Although Emergency Feedwater (EFW) actuated in automatic and fed both steam generators in response to low level, the Flow Control Valves (FCV) did not respond as expected (Ref 5.15).

The FCVs are designed to open based upon wide range steam generator level. Each FCV has an independent control circuit which includes an independent steam generator level indicator.

A review of steam generator level trends show a sudden spike downward in wide range steam generator level indications to less than 45% in less than 2 seconds, then increased to 74% in less than 2 seconds. The magnitude of the drop in indicated steam generator level is unrealistic (even accounting for the collapse of two phase flow in the tube bundle region), and the rapid recovery in indicated steam generator level is not supported by the quantity of feedwater that was being supplied to the steam generator. Rather, the rapid spike downward in indicated steam generator level was the result of a significant but brief loss of static and

dynamic equilibrium internal to the steam generator following a reactor trip, and is not indicative of actual steam generator level. Once these dynamic conditions stabilized, the indication was restored to accurately indicating actual steam generator level. The narrow range instruments, which initiate the Emergency Feed Actuation Signal (EFAS), were not affected by this dynamic equilibrium phenomenon.

Thus, the FCV control circuits received a low level input without a corresponding EFAS. The result was that although the EFW system did successfully feed the Steam Generators when an EFAS was received later in transient, the FCVs did not respond as expected.

This condition is unrelated to EPU. The condition was evaluated and determined to be acceptable for continued operation, and the EFW system remains operable. However, the condition was entered into the site Corrective Action program to determine if any additional action is required (Ref 5.15).

CONCLUSION:

The integrated plant control systems operated satisfactorily in automatic to stabilize the plant post trip. All safety systems responded as designed. Although one element of EFW response was not expected, the response was consistent with the system design. EFW responded acceptably to meet its safety function to restore and maintain steam generator level.

3.5 Load Changes Test (Control Systems Checkout)

PURPOSE:

The purpose of this test was to demonstrate that the integrated plant control systems operate satisfactorily in automatic to maintain plant parameters within specific limits.

BACKGROUND:

The Load Changes Test was planned to be performed coincident with Turbine Valve Testing scheduled for the fall of 2005, outside of the electric grid Summer Reliability window. However, prior to performance of this test, a normal plant shutdown was performed in response to Hurricane Katrina on August 28, 2005, followed by a normal plant startup to 100% RTP on September 13, 2005. The Load Changes Test was rescheduled for December 7, 2005; however, a manual trip of the reactor was performed on 11 November, at 2034 due to a total loss of Circulating Water (CW). A review of systems response to these two events was utilized in lieu of the Load Changes Test to demonstrate that the integrated plant

control systems operate satisfactorily in automatic to maintain plant parameters within specified limits in response to plant transients.

METHOD:

The Load Changes Test was planned to commence at steady state, 100% RTP. The planned Load Changes Test (100% to 90% to 95%) would observe the proper operation of the following systems during a plant power maneuver of up to 0.5%/min (~6 MW/min), from 100% to 90% and back to 95% RTP:

Automatic
- Feedwater Control System (FWCS)
- Feedwater Heater Drains (FHD) System
- Pressurizer Level Control System (PLCS)
- Pressurizer Pressure Control System (PPCS)
- Atmospheric Dump Valves (ADV) Control

Manual
- Chemical Volume Control System (CVCS)
- Digital-electro-hydraulic (DEH, turbine controls)
- Control Element Drive Mechanism Control System (CEDMCS)

In lieu of the Load Changes Test, a review of systems response to a normal plant shutdown and startup, and a loss of Circulating Water manual trip were performed. In comparison, the normal plant shutdown and startup was performed from the same initial condition and in the same manner as the planned Load Changes Test. All systems were operated in automatic or manual as was planned for the Load Changes Test. The load change rate achieved during the shutdown and startup were comparable to that planned for the Load Changes Test.

The manual reactor trip transient further exercised the following control system features:

- FWCS – Reactor Trip Override
- Steam Bypass Control System (SBCS) Quick Open
- Atmospheric Dump Valves (ADV) actuation and modulation
- Emergency Feedwater System (EFW) actuation and modulation
- Reactor Protection System (RPS) manual trip
- Control Element Drive Mechanism System (CEDMCS) manual trip.
- Pressurizer Level Control System (PLCS) modulation
- Pressurizer Pressure Control System (PPCS) modulation
- Digital-electro-hydraulic (DEH) – Automatic Turbine Trip upon Reactor Trip
- Main Generator – Automatic Generator trip upon Turbine Trip
- Electric Distribution – Automatic swap of house loads to offsite power (Startup Transformers) upon a Generator Trip

RESULTS:

The results of the plant shutdown and subsequent startup, and the loss of Circulating Water Plant trip are discussed separately above.

CONCLUSION:

The plant shutdown and startup in response to Hurricane Katrina demonstrated that the integrated plant control systems operate satisfactorily in automatic to maintain plant parameters within specific limits under conditions comparable to, and over a wider range of power levels than that planned for the Load Changes Test. Further, the loss of Circulating Water manual trip demonstrated the proper operation of a broader range of automatic system responses under transient conditions more extensive and rigorous than that planned for the Load Changes Test.

Thus, the plant shutdown and startup discussed above, combined with the loss of Circulating Water manual trip, satisfy the objectives of the Load Changes Test, and have adequately demonstrated that that the integrated plant control systems will operate satisfactorily in automatic to maintain plant parameters within specific limits under diverse transient conditions.

4.0 CONCLUSIONS

The completed Extended Power Uprate (EPU) Test program has demonstrated that the analyses, modifications, and adjustments necessary for EPU have been properly performed. Additional data has been collected to benchmark the analysis against the actual integrated performance of the plant. The results of testing performed during Power Ascension have been reviewed by the Onsite Safety Review Committee.

The results of the testing and equipment performance data gathering have demonstrated acceptable continued plant operation at the uprate power level of 3716 MWt.

5.0 REFERENCES

5.1 Waterford 3 Technical Specifications

5.2 Waterford 3 Cycle 14 Core Operating Limits Report (COLR)

5.3 Waterford 3 Final Safety Analysis Report (FSAR)

5.4 Waterford 3 Cycle 14 Reload Analysis Report

5.5 W3F1-2004-0004, Supplemental Information, Extended Power Uprate – Startup Testing

5.6 W3F1-2005-0067, Startup and Power Escalation Report for Cycle 14 Waterford Steam Electric Station, Unit 3

5.7 CR-WF3-2005-2799, SBCS Controlling RCS Temp High

5.8 Waterford-3 Cycle 14 Startup Test Predictions, Enclosure 1 to Westinghouse NF-WTFD-05-23.

5.9 CR-WF3-2005-2747, RC ITE0115 Out of Tolerance

5.10 CR-WF3-2005-2831, High Pressure Turbine Blades Design Error

5.11 CR-WF3-2005-2895, SDT 2B High Level

5.12 CR-WF3-2005-2868, Main Turbine Governor Valve Oscillations

5.13 CR-WF3-2005-3337, Elevated Vibration Startup Feedwater Reg Valves A&B, and MS-326A.

5.14 Waterford 3 Event Notification #42138, 4-hr Non-Emergency Report

5.15 CR-WF3-2005-04598, EFW Post-trip response

5.16 CR-WF3-2005-03685, Steam Generator Moisture Carryover Results