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Waterford 3

W3F1-2004-0004

January 29, 2004

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

SUBJECT: Supplemental Information
Extended Power Uprate – Power Ascension Testing
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCE: Entergy Letter dated November 13, 2003, "License Amendment Request NPF-38-249 Extended Power Uprate"

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) submitted an Extended Power Uprate (EPU) license amendment request for Waterford Steam Electric Station, Unit 3 (Waterford 3). During discussions with the NRC staff on December 18, 2003, Entergy agreed to submit additional information regarding the planned power ascension testing program as prescribed in draft Standard Review Plan 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs." This letter transmits this EPU power ascension testing information. As recommended by the NRC staff, the information is presented in a format and level of detail similar to that used by Vermont Yankee in letter BVY 03-98, dated October 28, 2003.

- Attachment 1 contains information regarding the planned modifications and associated testing activities.
- Attachment 2 provides a discussion of the aggregate impact of the modifications on the dynamic response of the plant.
- Attachment 3 contains the proposed EPU power ascension test plan.
- Attachment 4 compares original power ascension testing to the planned EPU power ascension testing.
- Attachment 5 provides the justification for exception to large transient testing.

The information provided by these attachments provides supplemental information to clarify information previously provided in Attachment 5 to the referenced letter. Therefore, the No Significant Hazards Consideration presented in the referenced letter is not changed by the information contained here in.

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This letter contains no new commitments. In section 2.10, of Attachment 5 of the referenced letter, Entergy committed to perform an EPU power ascension test program following refueling outage 13 in the spring of 2005. This commitment is unchanged by the information presented here in. The modifications and testing activities described here in are planned activities and, upon further evaluation, may be supplemented, modified, or deleted and therefore do not constitute commitments to be implemented specifically as described.

Entergy understands that the referenced EPU license amendment request will be considered complete and its review officially started upon NRC staff acceptance of this supplemental information. Therefore, Entergy requests approval of the referenced amendment within 12 months of the start of the official review. Once approved, the amendment will be implemented during restart from refueling outage 13 in the spring of 2005 and unit operation at the increased power level will occur in cycle14.

If you have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 29, 2004.

Sincerely,

J. R. Douet
General Manager, Plant Operations
Waterford Steam Electric Station, Unit 3

JRD/DBM/cbh

Attachments:
1. Extended Power Uprate Modifications and Testing
2. Aggregate Impact of Extended Power Uprate Modifications
3. Extended Power Uprate Test Plan
4. Comparison of Original Power Ascension Testing to Planned Extended Power Uprate Testing
5. Justification for Exception of Large Transient Testing

cc: Mr. Bruce S. Mallett
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Attachment 1

W3F1-2004-0004

Extended Power Uprate Modifications and Testing

Extended Power Uprate Modifications and Testing

The following table lists the planned modifications and prescribed acceptance testing necessary to support extended power uprate (EPU) for Waterford Steam Electric Station, Unit 3 (Waterford 3). These modifications, except where noted, will be implemented during refueling outage 13 (RF13) scheduled for the Spring of 2005. The following modifications constitute planned actions on the part of Waterford 3. Further evaluations may identify the need for additional modifications and tests or eliminate the need for some modifications and tests. As such, this list is not a formal commitment to implement the modifications and testing exactly as planned. Additionally, construction, installation, and/or pre-operational testing for each modification will be performed in accordance with the plant design process procedures and these tests are not listed herein. The tests listed are the final acceptance tests that will demonstrate the modifications will perform their design function and integrate appropriately with the existing plant.

Waterford 3 Extended Power Uprate Modifications/Testing		
System/Component	**Modification Description**	**Testing**
Plant Protection System	Reduce the low steam generator pressure trip bistable and annunciator setpoints.	Technical specification required channel calibration and functional test.
Atmospheric Dump Valves	Replace atmospheric dump valve controllers.	Channel calibration and functional test. This will include testing at the remote shutdown panel as appropriate.
Main Turbine	Replace the high pressure turbine steam path consisting of a new high pressure turbine rotor with all-reaction blading, a new inner cylinder with stationary blading, a new inlet flow guide, and steam sealing components.	• Baseline performance (pre-modification) test to document current electrical output. • Main turbine overspeed test. • Monitor turbine vibration. • Validate the core operating limits supervisory system (COLSS) constants used to determine plant power based on turbine first stage pressure. • Post-installation performance test to determine net change in electrical output as required.
Moisture Separator Reheater (MSR)	Provide additional shell side relief valve capacity.	Relief valve setpoint and capacity certified by vendor prior to installation.
Main Generator	• Rewind generator stator. • Modify step iron to increase cooling • Install a new stator cooling water alkalizer skid.	• Necessary electrical tests to demonstrate main generator can operate within its original capability curve. • Chemistry monitoring of stator cooling water • Monitor main generator vibration and support systems during power accession. • Monitor isophase bus duct temperatures.
Main Transformer	• Replace main transformer 'A'. • Upgrade cooling on main transformer 'B'.	• Test oil samples for degradation. • Monitor oil temperatures. • Periodic survey of 230 KV connection temperatures.

Waterford 3 Extended Power Uprate Modifications/Testing		
System/Component	**Modification Description**	**Testing**
Generator Output Breakers	Replace both existing generator output oil circuit breakers with higher capacity gas circuit breakers. (generator output breaker 'B' was replaced and tested during RF12 in the Fall of 2003).	• Electrical tests to demonstrate breakers will operate as designed. • Verify proper operation of trip circuits. • Verify proper operation of all auxiliary contacts. • Perform AC and DC acceptance tests. • Perform speed test to ensure proper input to synchronization check circuit. • Verify calibration of synchronization check circuit.
Feedwater Heater Drain System	Upgrade/replace normal and/or alternate level control valves as required.	• Normal level control valves – Verify heater level is maintained within operating band. • Alternate level control valves – Ensure seat leakage and valve stroke time is within design limits.
Main Condenser	Additional support staking of the main condenser tubes to minimize effects of flow induced vibration.	Monitor secondary chemistry.
Plant Control Systems and Instrumentation	• Pressurizer level controls - Revise reactor coolant system (RCS) T_{AVG} vs. pressurizer level program to accommodate lower RCS operating temperature. • Feedwater controls – adjustments to accommodate new process operating conditions • Steam bypass controls – adjustments to accommodate new process operating conditions. • Qualified safety parameter display system (QSPDS) – modify electronics to account for process range changes. • Feedwater/main steam instrumentation – re-span and change color banding on required instruments.	• Perform load change testing to verify automatic operation of the various control systems. • Collect plant data and confirm performance as expected. • Channel calibration and functional test.

Attachment 2

W3F1-2004-0004

Aggregate Impact of Extended Power Uprate Modifications

Waterford 3 Extended Power Uprate Modifications/Testing
Aggregate Impact on Dynamic Plant Response

The modifications listed in the table in Attachment 1 were reviewed to ensure the aggregate impact of the modifications do not adversely impact the dynamic response of the plant to anticipated initiating events. The majority of the modifications listed are required to allow operation at the new process conditions imposed by the extended power uprate (EPU). These modifications will not change the design functions of the equipment or the method of performing or controlling the function. The basis for this conclusion is discussed below.

The change to the low steam generator pressure trip setpoint (764 psia to 662 psia) is required to provide operating margin following EPU. Since the EPU proposed design will maintain the reactor coolant system (RCS) nominal hot leg temperature at 601°F, operating steam generator pressures would be close to the current low pressure trip setpoint. This could potentially cause spurious trips. The proposed low steam generator trip setpoint was incorporated into all applicable EPU safety analyses with acceptable results. The design function of this plant protection system setpoint, as discussed in Technical Specification Bases 2.2.1, will remain the same and the system response following this trip or another event initiator will not change as a result of this setpoint change.

Small break loss of coolant accident (SBLOCA) analysis for EPU requires crediting the safety-related atmospheric dump valves (ADVs) at a lower opening pressure setpoint when reactor thermal power is greater than 70% rated thermal power (RTP) to ensure 10CFR50.46 criteria are met. Due to the design characteristics of the current ADV controls, the lowering of the ADV setpoint with the current controls could cause the ADV to actuate prior to the steam bypass control system (SBCS) should a load rejection event occur. Consequently, the ADV controller will be replaced with a more accurate controller so that the SBCS will actuate prior to the ADV to control the steam release should a load rejection event occur. This modification does not result in a new system interaction. The current SBLOCA analysis credits the main steam safety valves (MSSVs) to provide the secondary pressure control. This function will still be performed by the main steam system but EPU will also credit the safety-related ADV for performing this function. The ADV has adequate capacity to perform this function. Therefore, the design function of the main steam system will remain the same following implementation of this modification.

The high-pressure turbine steam path must be replaced in order to accept the higher steam flows generated at the higher power level. The new steam path will change normal turbine control operation from a 'sequential' valve operation to a 'single' valve operation. This change will not modify or impact the turbine throttle and governor valves. Also, change to the current turbine control system is not required to make this transition. The current control system allows for 'single' valve operation and this mode of operation is currently used during plant startup and turbine valve testing. Therefore both the design function of the high pressure turbine and the system response to event initiators will not change as a result of this modification.

The main generator modifications are required due to the normal degradation and corrosion experienced in the stator cooling water coils. The stator rewind and core step iron modifications will restore the main generator to its original capability which will be needed to operate at the higher electrical output. A new alkalizer skid will be retrofitted into the existing generator stator cooling water system. The addition of the stator water alkalizer skid will enhance the reliability of the main generator by minimizing corrosion to the stator cooling water coils. These generator modifications do not change the design function of the main generator nor will a new system interaction be created. System response to a turbine runback/setback actuation will also not change as a result of these main generator modifications.

The changes to the plant control systems are required to ensure the plant, at the new EPU operating conditions, will be maintained at desired operating bands during normal operations and will stabilize the plant during minor load changes and load rejection events. These adjustments do not change the design functions of the equipment or the method of performing or controlling the function. Further, the computer code used to simulate the plant control systems and determine the proposed changes to control system parameters has been previously used in developing the existing control system settings and benchmarked against actual plant events. Therefore, these adjustments will not result in a significant change to the plant's dynamic response to anticipated initiating events.

The purpose of the modifications to moisture separator reheater relief valves, main transformers, generator output breakers, selected feedwater heater control valves, and main condenser are to upgrade the components to accept the EPU operating conditions. These modifications will not result in a significant change to the plant's dynamic response to anticipated initiating events.

Attachment 3

W3F1-2004-0004

Extended Power Uprate Test Plan

Waterford 3 Extended Power Uprate Power Ascension Test Plan																									
Test / Modification	**Test Description [1]**	**Prior to Startup**	**Rated Thermal Power – % of 3716 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**	**90**	**92.5**	**95.0**	**97.5**	**100**
Plant Protection System - low steam generator pressure setpoint	Attachment 1	X																							
Atmospheric Dump Valve - controller modification	Attachment 1	X																							
Main Turbine - high pressure turbine steam path replacement	Overspeed Test					X																			
	Attachments 1 & 4	X	X		X		X		X		X		X		X		X		X		X	X	X	X	X
Moisture Separator Reheater - shell relief valve upgrade	Attachment 1	X																							
Main Generator - stator rewind	Attachment 1	X					X						X					X				X	X	X	X
Main Generator - stator coil water alkalizer modification	Attachment 1	X																							
Main Transformer 'A' - replacement	Attachment 1	X				X	X		X		X		X		X		X		X		X	X	X	X	X

[1] Line items may have multiple tests. Each test will not necessarily be performed at every power level indicated.

Waterford 3 Extended Power Uprate Power Ascension Test Plan																									
Test / Modification	**Test Description [1]**	**Prior to Startup**	**Rated Thermal Power – % of 3716 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**	**90**	**92.5**	**95.0**	**97.5**	**100**
Main Transformer 'B' - cooling modification	Attachment 1	X				X	X		X		X		X		X		X		X		X	X	X	X	X
Main Generator - output breaker replacement	Attachment 1	X							Between 30%-50%																
Feedwater Heater - drain level control valve upgrades	Attachment 1	X					X						X						X			X			X
Main Condenser - tube staking	Attachment 1	X		X																					X
Plant Control Systems and Instrumentation – adjustments	Attachment 1	X	X		X		X		X		X		X		X		X		X		X	X	X	X	X
Main Steam and Feedwater Piping Flow Induced Vibration	Attachment 4	X																							X
Steam Generator	Measure moisture carryover	X																							X
Emergency Diesel Generator	Measure fuel oil consumption rate [2]	X																							

[2] As committed to in Section 2.5.8.1 of the EPU Report. (A26623)

Waterford 3 Extended Power Uprate Power Ascension Test Plan

Test / Modification	Test Description [1]	Prior to Startup	Rated Thermal Power – % of 3716 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	90	92.5	95.0	97.5	100
Nuclear Steam Supply System (NSSS) Data Record	Data Collection						X		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	X	X
Nuclear and Thermal Power Calibration	Normal startup calibrations.						X										X					X	X	X	X
NSSS Calorimetric	Verify accuracy of COLSS		X																			X	X	X	X
Reactor Coolant System (RCS) Calorimetric Flow Measurement	Normal startup test – RCS flow		X														X					X	X	X	X
Linear Power Subchannel Calibration	Data Collection - Adjust excore linear power to match incore detectors if needed	X					X		X		X		X		X		X								
Process Variable Intercomparison	Attachment 4																					X			X
Radiation Surveys	Perform radiation surveys																					X			X

Waterford 3 Extended Power Uprate Power Ascension Test Plan																									
Test / Modification	**Test Description [1]**	**Prior to Startup**	**Rated Thermal Power – % of 3716 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**	**90**	**92.5**	**95.0**	**97.5**	**100**
Core Performance Record	Evaluates SAM and BPPCi and compares various power distribution parameters to predictions						X		X		X		X		X		X								
CPC/COLSS Verification	Evaluate process noise for CPC																					X			X
Variable Tavg	Measures ITC, MTC and PC		X																						X
Load Changes	5% ramp to verify control system response																						X		
Shape Annealing Matrix Measurement	Measures SCij and BPPCi and compared to CPCs						X		X		X		X		X		X								
Radial Peaking Factor Verification	Measure planar radial peaking and CEA shadow factors and calibrate COLSS and CPCs																X					X	X	X	X

Waterford 3 Extended Power Uprate Power Ascension Test Plan																								
Test / Modification	Test Description [1]	Prior to Startup	Rated Thermal Power – % of 3716 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	90	92.5	95.0	97.5	100
COLSS Power / Flow Verification Data Record	Determine proper constants for COLSS turbine power calculation		X		X		X		X		X		X		X		X		X		X	X	X	X	X
Adjustment of COLSS Secondary Pressure Loss Terms	Data Collection																					X			X
Balance of Plant Data Record	Data Collection						X		X		X		X		X		X		X		X	X	X	X	X
Thermal Expansion	Data Collection – Balance of piping walkdowns as necessary	X																							

<u>Acronyms used:</u>

BPPCi – Boundary Point Power Constants
CEA – Control Element Assembly
COLSS – Core Operating Limits Supervisory System
CPCs - Core Protection Calculators
ITC – Isothermal Temperature Coefficient
MTC – Moderator Temperature Coefficient
PC – Power Coefficient
QSPDS – Qualified Safety Parameter Display System
SAM – Shape Annealing Matrix

Attachment 4

W3F1-2004-0004

Comparison of Original Power Ascension Testing to
Planned Extended Power Uprate Testing

Comparison of Original Power Ascension Testing to Planned Extended Power Uprate Testing

In accordance with draft Standard Review Plan (SRP) 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," the licensee should provide a comparison of the proposed extended power uprate (EPU) testing program to the original power ascension test program performed during initial plant licensing. The scope of this comparison should include (1) all power ascension tests initially performed at a power level of equal to or greater than 80 percent of the original licensed thermal power level; and (2) initial power-ascension tests performed at lower power levels if the EPU would invalidate the test results. The following table presents the comparison of the planned Waterford Steam Electric Station, Unit 3 (Waterford 3) EPU power ascension test program with the original power ascension test program starting with low power physics testing conducted at hot zero power conditions continuing through 100% power testing.

The first column of the table (SU Test #) provides information that can be used to cross reference between the Startup (SU) Test number (e.g., SIT-TP-650), the section of the Cycle 1 Startup Test Report [1] (e.g., 5.0) where the testing and results are described, and the section(s) of the Final Safety Analysis Report (FSAR) (e.g., 14.2.12.3.10) related to the test.

The second column of the table (Test Description) provides the description/purpose of the original power ascension test taken from the Cycle 1 Startup Test Report. The third column of the table (Original Power Ascension Test Power Level) identifies the power level or range of power levels at which the original power ascension test was performed.

The fourth column (Test Plan for EPU) indicates whether all of, a part of, or testing similar to the original power ascension test need be performed as a result of the EPU. A 'yes' indicates that testing should be performed to revalidate plant performance as a result of EPU. The testing planned may or may not be the original power ascension test but will validate proper plant performance. A 'no' indicates that the testing need not be performed as a result of EPU but does not necessarily mean that such testing will not be performed following the refueling outage. (e.g., Low power physics testing need not be performed as a result of the EPU but it will be performed, as it normally is, following a refueling outage.) Attachment 3 identifies the testing planned as a result of EPU and the power levels at which testing is planned to be performed.

The fifth column (Evaluation/Justification) provides the basis for not performing the original power ascension test as a result of EPU. For the large transient tests, refer to Attachment 5 for this justification. In some cases, when testing will be performed, this column provides additional clarification on the testing to be performed.

[1] Louisiana Power and Light (i.e., Entergy) letter W3P85-3218 to the USNRC dated October 10, 1985 as amended by letter W3P85-0048 to the USNRC dated March 14, 1986.

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
SIT-TP-650 5.0 14.2.12.3.10, 14.2.12.3.11, 14.2.12.3.12, 14.2.12.3.13, & 14.2.12.3.14	**Low Power Physics Test** The Low Power Physics Test verified the physics parameters pertinent to the Waterford 3 reactor by comparing measured results to predicted values.	0%	No	The original low power physics test requirement is not changed by EPU. It is performed for each reload in accordance with approved plant procedures and Technical Specifications. This testing will be performed, as required by Technical Specifications, during refueling outage 13 in the Spring of 2005. However, there is no low power testing specifically required for EPU.
SIT-TP-701 6.7.1 NA	**Nuclear Steam Supply System (NSSS) Plant Data Record** This test provided a permanent baseline data record of plant parameter indications from zero power to full power operation, during steady state operation.	0% - 100%	Yes	The plant monitoring computer was used to provide this information with the exception of boron concentration. The original startup test baseline data remains valid. The EPU power ascension will be recorded on an upgraded plant monitoring computer with an expanded set of information that is readily available for review. This plant computer information will be collected and saved electronically. Boron concentration is measured and recorded as required per existing plant procedures and Technical Specifications to determine adequate shutdown margin.
SIT-TP-702 6.7.2 NA	**Transient Data Record** The transient data record established a plant baseline data record during the slow initial power increases of the plant. The data provides an overview of primary and	0% - 100%	Yes	The plant monitoring computer was used to acquire the information with the exception of data from the core protection calculator (%power, Tcold, DNBR, ASI), steam bypass control system (SBCS) master control demand, and SBCS valve demand. The core protection calculator (CPC) and SBCS information is

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
	secondary plant loads and operating conditions and how they change during power increases.			now available from the upgraded plant-monitoring computer. Information will be collected via the upgraded plant computer and saved electronically.
SIT-TP-704 6.1.1 14.2.12.3.27	**Reactor Coolant System (RCS) Delta-T Power Determination** This test determined the thermal output of the reactor by means of a primary system calorimetric. The power level calculated in this test was then used as the standard for calibrating the CPCs and the excore nuclear instrumentation.	5% - 20%	No	This manual (hand) calculation was performed to establish a "standard thermal power" for first time use in other power ascension test performed at the time of the original startup. The core operating limits supervisory system (COLSS) is currently used to calculate BSCAL which is used to calibrate CPCs and excore nuclear instrumentation. The manual calculation is no longer needed since baseline information from previous operating history is available.
SIT-TP-705 6.2.1 14.2.12.3.27	**Nuclear and Thermal Power Calibration** The nuclear and thermal power calibration test calibrated excore linear power, CPC thermal power, and CPC nuclear power to a standard measurement of core power.	0% - 100%	Yes	The power indications will be recalibrated to the new EPU 100% power conditions.
SIT-TP-707 6.7.5 14.2.12.3.29	**SBCS Capacity Checks** This test verified that the steam flow capacities of the two atmospheric dump valves (ADVs) and of the six turbine bypass valves (TBVs) were in accordance with design requirements and safety analysis assumptions.	60%	No	This test verified that the maximum capacity of a turbine bypass valve or atmospheric dump valve is less than that assumed in the most severe excess heat removal accident and that the minimum capacity of an atmospheric dump valve is sufficient to remove the assumed decay heat. The physical characteristics of the bypass valves and atmospheric dump valves are not being altered for EPU therefore; the physical capacity of the bypass valves and atmospheric dump valves is not changing. Extended Power Uprate design

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
				reviews have assured that the existing capacity of these valves is acceptable under EPU conditions, thus additional capacity testing is not required.
SIT-TP-708 6.7.6 NA	**Initial Turbine Startup** This test verified 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 and a baseline record of turbine operation was established.	0% - 100%	Yes	Verification of acceptable operation of the following will be performed as part of the high pressure turbine steam path replacement modification testing or as part of plant start-up. • Acceptable operation of the turbine lube oil pumps. • Acceptable operation of the throttle, governor, intercept and reheat stop valves. • Acceptable operation of the turbine trip due to low bearing oil pressure, low vacuum, or thrust bearing failure. • Acceptable operation of the overspeed trip at the turbine pedestal. • Acceptable operation of the "Test" turbine trip mechanism. • Acceptable operation of the mechanical overspeed turbine trip.
SIT-TP-709 6.1.2 14.2.12.3.27	**NSSS Calorimetric** The NSSS calorimetric power measurement provided an accurate determination of reactor power based on a secondary plant energy balance.	0% - 100%	Yes	The power measurements will be revised to the new EPU 100% power conditions. The methodologies will remain unaffected.
SIT-TP-710	**RCS Calorimetric Flow Measurement**	20% - 100%	Yes	The RCS flow measurement will be revised to the new EPU 100% power conditions in accordance with

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
6.5.1 14.2.12.3.2	This test determined an accurate value of the RCS flowrate; this measured flowrate was then used as the standard to which the COLSS and CPC calculated RCS flowrates were conservatively calibrated. A second purpose of this test was to recalibrate the COLSS and CPC thermal powers to secondary calorimetric power following adjustment of their respective flow rates. Finally, the test gathered data for use in the evaluation of the adequacy of the installed thermal power adjustment coefficients.			Technical Specifications. The methodologies will remain unaffected.
SIT-TP-711 6.2.3 14.2.12.3.28	**Linear Power Subchannel Calibration** This procedure adjusted the excore linear power subchannel amplifier gains so that the fractional power distribution as measured by the excore detectors was within 0.1% of that measured by the incore detectors. After completing the adjustments, the excore 200% linear calibrate potentiometers were reset to reflect the new amplifier gains. The second part of this procedure collected baseline data on all the amplifiers to be used for routine surveillances or replacement of amplifiers.	20% - 50%	Yes	Based on previous operating experience and the core reload design process, the linear subchannel calibration will be adjusted prior to startup. The methodologies used in previous cycles will remain unaffected.
SIT-TP-712	**Process Variable Intercomparison**	20% - 100%	Yes	This intercomparison will be performed above 3441

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
6.2.2 14.2.12.3.30	This test demonstrated that the inputs and appropriate outputs of the Plant Protection System (PPS), the CPCs, and the Plant Monitoring Computer (PMC) were in satisfactory agreement with one another. Permanent plant instrumentation (meters and recorders) were also included in the intercomparison.			MWt.
SIT-TP-713 6.7.11 14.2.12.3.16	**Chemistry** The objective of the chemistry testing was to verify proper operation of chemistry sampling and analysis equipment in addition to establishing the adequacy of procedural controls utilized to monitor and maintain chemistry and radiochemistry conditions within the Waterford 3 primary and secondary systems.	0% - 100%	No	EPU will have no effect on the operation of the chemistry sampling equipment. Current chemistry controls are adequate and will remain in place to maintain chemistry and radiochemistry conditions at the uprated conditions in the primary and secondary systems.
SIT-TP-714 6.2.4 14.2.12.3.40	**Vibration and Loose Parts Monitoring** This test established baseline data for all vibration, loose parts, and reactor core internals motion channels at various test plateaus and verified that the existing loose parts alarm setpoints are acceptable for power operation.	0% - 100%	No	In 1991, a modification was implemented to replace the sensors tested under SIT-TP-714. The new system takes automatic noise background baseline checks. The new system was tested after the modification was complete and was verified to perform satisfactorily. Periodic calibrations and functional checks of the system required by the Technical Requirements Manual have verified the system continues to work as designed.

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
				The EPU will have no impact on the dynamic characteristics of the primary coolant system. The primary flow does not change significantly as a function of power. The RCS is not being reconfigured as a result of the EPU, hence there is no change to RCS flow. The only changes to RCS flow that would occur would result from the slight bulk density change due to changes in RCS cold leg operating temperatures. Additionally, the impact of revised thermal, hydraulic, mechanical and pipe breaks input data due to power uprate was evaluated for the RVI components. The revised thermal input data reflects changes in temperature distribution associated with EPU. The revised hydraulic input in the form of hydraulic loads, moments and pressure differentials are consistent with the current fuel assembly design. The revised mechanical input in the form of weights, fuel spring loads and hold down ring loads are also consistent with the current fuel assembly design. Based on these analyses, there should be no impact on the vibration and loose parts monitoring system.
SIT-TP-715 6.7.3 14.2.12.3.15	**Biological Shield Effectiveness Survey** This test obtained baseline radiation levels in order to trend radiation level buildup with operation; to measure and document radiation levels in locations outside of the biological shield while at power; to establish the adequacy of the	5% - 100%	Yes	Selected dose rate surveys will be performed at the increased power level.

SU Test # SU Report Section # FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
	biological shield and to identify high-radiation zones.			
SIT-TP-716 6.4.1 14.2.12.3.27	**Core Performance Record** The core performance tests determined if the predicted CPC shape annealing matrix and boundary point power constants provided an acceptable synthesis of the core average axial power distribution until a measurement of these parameters was made at 50% rated thermal power. This screening was performed at the 20% test plateau only. The core performance tests also verified that the core design and construction were as expected by comparing various power distribution parameters to predictions.	20% - 100%	Yes	The cycle independent shape annealing matrix and boundary point power constants will be validated in accordance with current procedures and Technical Specifications. The original startup testing program performed this test at power levels above 70%. This was done because there was no previous operating experience calibrating the Waterford 3 CPC power distribution using the excore nuclear instruments. The excore nuclear instruments are now verified against the cycle independent shape annealing matrix (CISAM) between 20% and 70% power. Meeting the acceptance criteria at these power levels validates that the excore nuclear instruments will respond properly at power levels greater than 70% power. Therefore, testing above 70% power is not necessary for EPU startup testing.
SIT-TP-717 6.3.3 14.2.12.3.27	**CPC/COLSS Verification** The CPC/COLSS verification test verified the CPC and COLSS calculations of departure from nucleate boiling ratio (DNBR) and local power density (LPD). The test also collected input recordings used to evaluate the effects of process noise on the CPC system.	0% - 100%	Yes	Algorithms in CPC/COLSS are unchanged for EPU. Process noise in the CPC system will be evaluated at 100%.

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
SIT-TP-718 6.4.2 14.2.12.3.26	**Variable Tavg** The variable Tavg test measured the isothermal temperature coefficient (ITC), moderator temperature coefficient (MTC) and power coefficient (PC).	50% - 100%	Yes	Testing to be performed prior to exceeding 40 effective full power days in accordance with Technical Specifications.
SIT-TP-721 6.2.5 & 6.6.2 14.2.12.3.31 & 14.2.12.3.39	**Load Changes (Control Systems Checkout)** This test demonstrated that the integrated plant control systems (steam bypass control system (SBCS), feedwater control system (FWCS), reactor regulating system (RRS), pressurizer level control system (PLCS), pressurizer pressure control system (PPCS), digital-electro-hydraulic (DEH) system, and control element drive mechanism control system (CEDMCS)) operated satisfactorily in automatic to maintain plant parameters within specific limits.	50% - 100%	Yes	A test to gather data as part of EPU power ascension will be performed. This will be integrated with EPU modification tests on plant control systems to verify proper operation under EPU conditions.
SIT-TP-723 6.3.6 14.2.12.3.28	**Shape Annealing Matrix Measurement** This test determined the relationship between the excore detectors and the incore power distribution. Specifically, the Shape Annealing Matrix (SAM) elements (i.e. SCij) and the Boundary Point Power Correlation Coefficients (BPPCi.) were	20% - 50%	Yes	The cycle independent shape annealing matrix and boundary point power constants will be validated in accordance with current procedures and Technical Specifications.

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
	measured and compared to the corresponding values used by the CPCs to determine if the CPC values were appropriate.			
SIT-TP-724 6.3.5 14.2.12.3.28	**Temperature Decalibration Verification** The temperature decalibration verification test measured the effect of changes to the excore signals due to changes in cold leg temperature and verified that the temperature shadowing factors installed in the CPC's were adequate.	50%	No	Algorithms contained within CPCs are unchanged. The "update" algorithm within CPCs accommodates for changes in cold leg temperature and applies a conservative temperature shadowing factor. Therefore, no special test is required.
SIT-TP-725 6.3.4 14.2.12.3.28	**Radial Peaking Factor Verification** The planar radial peaking factor/CEA shadowing factor test obtained a direct measurement of these parameters for various CEA insertion configurations. Based upon these measurements, additional assurance of correct core loading was obtained. The COLSS and the CPCs were also calibrated such that they accounted for the measured values of these parameters.	50%	Yes	The planar radial peaking factor and control element assembly shadowing factor will be confirmed in accordance with Technical Specifications.
SIT-TP-726 6.6.1	**Remote Reactor Trip with Subsequent Remote Cooldown** This test demonstrated the following:	20%	No	Reference Attachment 5.

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
14.2.12.3.33	1) There was equipment provided at appropriate locations outside the control room which had the capability to trip the reactor. 2) There were adequate instrumentation and controls outside the control room to maintain the plant in a hot standby condition. 3) Cold shutdown of the reactor from outside the control room was achievable. 4) The plant operating procedures used in performing the remote shutdown and cooldown were sufficiently clear and comprehensive, and the operating personnel were familiar with their application. 5) The minimum required shift complement was sufficient to perform the actions required for the remote shutdown and the maintenance of hot standby. No design deficiencies or potential hazards to plant equipment or personnel existed during a remote reactor trip and subsequent remote plant cooldown.			
SIT-TP-727 6.6.4	**80% Total Loss of Flow / Natural Circulation** The Loss of Flow test demonstrated that	80%	No	Reference Attachment 5.

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
14.2.12.3.34	the dynamic response of the plant to a total loss of forced reactor coolant flow following sustained power operation is in accordance with design and that stable natural circulation can be established to maintain adequate core cooling.			
SIT-TP-728 6.6.3 14.2.12.3.35 & 14.2.12.3.41	**Loss of Offsite Power Trip** This test demonstrated plant performance under a total loss of AC power. This test verified that the reactor can be shutdown and hot standby conditions can be achieved and maintained using engineered safety features (ESF) power (4160 volt emergency diesel generators). Additionally, by simulating a total loss of onsite AC power, this test demonstrated the ability to remove decay heat with natural circulation flow in the reactor coolant system and with secondary feed from the steam-driven emergency feedwater pump.	20%	No	Reference Attachment 5.
SIT-TP-735 6.2.6 14.2.12.3.3	**Incore Detector Signal Verification** 1) This test verified proper operation and signal processing of the incore detector signals to assure accurate power distribution calculations.	20% - 100%	No	The in-core instrumentation system is used to confirm core power distribution, perform periodic calibrations of the excore flux measurement system, and provide inputs to COLSS. The fixed incore detector system used at Waterford 3 is unchanged by EPU. The range of the equipment is sufficient to encompass the range at EPU conditions. Additionally, the same design

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
				rhodium detector is used in similar reactor designs that operate at near EPU power levels. The detectors are replaced and tested periodically to ensure that accuracy is not affected by the depletion of the rhodium. Therefore no specific testing is required as a result of EPU.
SIT-TP-739 6.3.1 NA	**COLSS Power / Flow Verification Data Record** This test collected data at 0% power for use in adjusting the COLSS delta-T power algorithm for zero-power conditions. The test was also used to verify that the correct values of COLSS constants affecting power and flow were inserted at 0%. Additionally, the test collected data at each 10% power plateau between 20% and 100% to be used in determining the proper constants for the COLSS calibrated turbine power calculation.	0% - 100%	Yes	Zero power constants will be determined as part of startup testing. Additionally, data will be collected to confirm or adjust "block G" constant calculations to correlate turbine first stage pressure with reactor power.
SIT-TP-740 6.6.5 14.2.12.3.37	**100% Turbine Trip** This test demonstrated that the plant design is adequate to respond to a 100% power turbine trip and that plant systems respond in accordance with design. The data collected on plant response was used to verify computer code predictions, which were used for modeling plant	100%	No	Reference Attachment 5.

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
	transients.			
SIT-TP-741 6.3.2 NA	**Adjustment of COLSS Secondary Pressure Loss Terms** This test tuned the COLSS algorithms which calculate steam generator pressure and feedwater pressure. This test collected live data for both the dependent (feedwater and steam generator pressures) and the independent (steam header pressure and steam flow) parameters used in the algorithms modeling the secondary pressure losses. This data was then used to determine the constants to be implemented into the COLSS algorithms.	0% - 100%	Yes	Appropriate data will be collected to determine constants for the COLSS pressure loss algorithm.
SIT-TP-743 6.7.4 14.2.12.3.32	**Ventilation Capability** This test verified that various heating, ventilation and air conditioning (HVAC) systems for the containment, annulus, areas housing engineered safety features (ESF) and areas housing ESF support systems were able to maintain design temperatures while the plant was operated at or near specified power levels (50% and 100%), and during a plant cooldown and a loss of offsite power condition.	0% - 100%	No	For areas inside containment and the annulus, EPU does not propose physical modifications that would add additional heat loads, propose adding, removing or replacing piping insulation, or increase demands on operating equipment during a plant cooldown or a loss of offsite power. The initial power ascension ventilation testing was conducted under the original plant design conditions. These conditions included an RCS hot leg temperature of 611°F with a main steam header pressure of 900 psia. The current and proposed EPU nominal operating RCS hot leg temperature is 601°F and the EPU

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
				expected main steam pressure will be 810 psia. For outside containment areas that house ESF and ESF support systems, area heat loads will remain essentially unchanged with the exception of the shutdown cooling heat exchanger (SDCHX) rooms. The increased decay heat from EPU conditions during normal shutdown and accident conditions will increase the component cooling water piping heat load contribution into the SDCHX rooms. The impact of this increased heat load is being evaluated (reference commitment for Section 2.5.5.3). If design margins on the HVAC equipment in the SDCHX rooms are inadequate, modifications will be performed as necessary to ensure the design room temperatures are maintained. If SDCHX room HVAC modifications are warranted, acceptance testing will be performed to demonstrate room temperatures will be maintained below the design limits.
SIT-TP-748 6.7.7 NA	**Balance of Plant Data Record** This test collected data relative to secondary plant systems and components in order to establish an initial data base to be used for future performance comparisons and analyses.	0% - 100%	Yes	The plant monitoring computer was used to acquire the information with the exception of feedwater heater drain line flow and moisture separator reheater (MSR) shell drain tank drain line flow. Information will be collected on an upgraded plant monitoring computer and saved electronically. Additionally, feedwater heater drain line flow and MSR shell drain tank drain line flow for the power ascension will also be recorded.
SIT-TP-749 **SIT-TP-751**	**Reactor Power Cutback System (RPCS) Loss of Load Tests**	NA	No	Reference Attachment 5.

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
SIT-TP-752 N/A 14.2.12.3.38	The purpose of this test, from a nominal 50% power, was to demonstrate that on a loss of load there would be no reactor power cutback, no engineered safety features actuation system (ESFAS) actuation, no reactor trip, no lifting of primary or secondary code safeties, and that the control systems respond properly to the transient by stabilizing reactor power at the desired power. The purpose of this test, from a nominal 80% and 100% power, was to demonstrate that on a loss of load, that preselected control element assembly subgroup insertion occurs, that there is no ESFAS actuation, no reactor trip, no lifting of primary or secondary code safeties, and that the control systems respond properly to the transient by stabilizing reactor power at the desired power.			The non-safety related RPCS is not credited for the mitigation of any design basis accidents and was not specifically tested during the initial power ascension testing program.
SIT-TP-750 **SIT-TP-753** NA 14.2.12.3.42	**RPCS Loss of Feedwater Pump Tests** The purpose of this test, from a nominal 70% power, was to demonstrate that on a loss of one feedwater pump there would be no reactor power cutback, no ESFAS actuation, no reactor trip, no lifting of primary or secondary code safeties, and	NA	No	Reference Attachment 5. The non-safety related RPCS is not credited for the mitigation of any design basis accidents and was not specifically tested during the initial power ascension testing program.

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
	that the control systems respond properly to the transient by stabilizing reactor power at the desired power. The purpose of this test, from a nominal 100% power, was to demonstrate that on a loss of feedwater that preselected control element assembly subgroup insertion occurs, no ESFAS actuation, no reactor trip, no lifting of primary or secondary code safeties, and that the control systems respond properly to the transient by stabilizing reactor power at the desired power.			
SIT-TP-755 6.5.2 14.2.12.3.25	**Natural Circulation Demonstration** This test demonstrated that natural circulation flow conditions and heat removal capability are in accordance with design.	80%	No	Reference Attachment 5.
SIT-TP-900 6.7.9 14.2.12.3.17	**Pipe Whip Restraint Measurements** This test verified by measurement and/or observation that all pipe whip restraints, both soft ('U' - bar type) and hard (rigid restraint), cleared all piping, piping insulation and piping components during cold and normal operating condition. The clearances between pipe and whip	20% - 100%	No	The acceptance criteria for whip restraints have been satisfied for all piping for which analyses have been completed. As committed to in Section 2.2.2.2 of the EPU Report, (reference commitment A26603) Entergy will insure that all remaining ASME Class 2 lines (i.e., main steam and feedwater lines outside containment) will continue to meet current design requirements following EPU. If analysis shows that these lines fail to meet the current design requirements in their current

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
	restraints have to be within specified tolerances, in order to perform their intended design function.			configuration they will be redesigned and modified to meet the current design requirements for EPU prior to operating at EPU conditions.
SPO-99P-003 3.3.5 14.2.12.3.17	**Thermal Expansion** This test verified that piping and component expansions are free, unrestrained and within tolerance (during plant heat-up and normal operation) as predicted by analysis.	20% - 100%	Yes	Since the original thermal expansion test was performed, the nominal hot leg temperature has been reduced from 611°F to 601°F and the nominal hot full power cold leg temperature has been reduced from 553°F to 543°F. A thermal expansion evaluation was conservatively performed using the pre-EPU temperature conditions. Effects of revised or new thermal transient and /or thermal stratification transients were considered as applicable in this evaluation. Results of the evaluation indicate that all pipe movements are lower than those for which the plant was qualified or sufficient clearances exist to provide unrestrained pipe thermal movement based on information provided in the analysis of record except for a small portion of piping. A plant walk down of the charging and letdown piping was performed during refueling outage 12 in the Fall of 2003 and confirmed adequate clearance existed to accommodate the expected pipe movement. Additional plant walk downs are planned for refueling outage 13 to confirm the remaining piping has adequate clearance to accommodate the expected pipe movement prior to EPU.
SPO-99P-004	**Level 2 Piping Vibration Testing**		Yes	Main steam and feedwater system flows are increasing

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
6.7.8	Original vibration testing was done in accordance with SPO-99P-001, Level 1 and Level 2 piping vibration testing. The purpose of this subsequent test was to verify by measurement and/or observation that vibration amplitudes were acceptable for piping systems or portions thereof whose configuration and/or support locations were changed after completion of vibration testing performed in accordance with procedure SPO-99P-001			for EPU therefore Entergy will obtain vibration measurements at pre-determined locations on these systems. Process flows are not expected to increase for the remainder of the primary plant systems, therefore testing will not be required. Note: The purpose of SPO-99P-001 was to verify by observation (Level 1) and by measurements (Level 2) that the amplitudes of vibration piping within pre-selected systems were acceptable.
STP-36 6.7.10 14.2.12.3.17 14.2.12.2.95	**Inspection of Mechanical Snubbers and Spring Supports** The purpose of this test was to verify by measurements and/or observation that mechanical snubbers and spring supports were properly installed and responding correctly to design criteria during plant heat-up, cooldown and normal operating conditions.		No	Piping and support evaluations have been completed for EPU (except for supports on main steam and feedwater piping outside containment) and they do not require any modification based on the analysis results. Baseline data obtained prior to plant start-up and/or later during a snubber reduction program is acceptable and no further measurements and/or visual examination of the snubbers and springs are required for EPU. As committed to in Section 2.2.2.2 of the EPU Report, (reference commitment A26603) Entergy will insure that all remaining ASME Class 2 lines (i.e., main steam and feedwater lines outside containment) will continue to meet current design requirements. If analysis shows that these systems fail to meet the current design requirements in their current configuration they will be redesigned and modified to meet the current design requirements for EPU prior to operating at EPU

SU Test # / SU Report Section # / FSAR Section #	Test Description	Original Power Ascension Test Power Level	Test Plan for EPU (Yes/No)	Evaluation / Justification
				conditions. Therefore no specific startup test is necessary to assure whip restraint clearances.

Attachment 5

W3F1-2004-0004

Justification for Exception to Large Transient Testing

Justification for Exception to Large Transient Testing

Entergy Operations, Inc. (Entergy) has reviewed the recommendations of draft Standard Review Plan (SRP) 14.2.1 for extended power uprate (EPU) testing programs. As a result of this review and a review of the original Waterford Steam Electric Station, Unit 3 (Waterford 3) startup test program, Entergy concludes that only the load change test discussed in Section 6.6, "Transient Testing," of the Cycle 1 Startup Test Report[1] need be performed as part of EPU power ascension tests. This attachment discusses the justification for not performing the remaining transient tests discussed in Section 6.6 of the Cycle 1 Startup Test Report.

The following transient tests were proposed for the initial power ascension test program.

Test #	Cycle 1 Transient Testing	EPU Test
SIT-TP-721	Load Changes	Yes
SIT-TP-726	Remote Reactor Trip With Subsequent Remote Cooldown	No
SIT-TP-727	80% Total Loss of Flow Test/Natural Circulation	No
SIT-TP-728	Loss of Offsite Power Trip	No
SIT-TP-740	100% Turbine Trip	No
SIT-TP-749, 750, 751, 752, 753	Reactor Power Cutback System (RPCS) Loss of Load and Loss of Feedwater Pump Testing.	No
SIT-TP-755	Natural Circulation Demonstration	No

The 8% Waterford 3 Extended Power Uprate developed its operating point to correspond to a nominal HFP Thot value of 601°F, which is approximately the same as for current licensed conditions and less than the nominal Thot value with which Waterford 3 operated until 1992. There are no major changes to the nuclear steam supply system due to EPU. The major plant modifications as noted in Attachment 1 are to the turbine-generator, main transformers and switching station. As a result, Entergy believes that of these transient tests, only a load change test should be re-performed as a result of EPU. Re-performing the remaining transient tests for EPU is not necessary. If performed, such tests would not confirm any new or significant aspect of performance which has not already been demonstrated by previous operating experience or is routinely demonstrated through plant operation.

Unlike during initial startup, when plant data was not available for compiling analytical models for CESEC and other computer codes, Waterford 3 operational data has been used in the construction of the CENTS model for EPU which is being used for FSAR Chapter 15 safety analyses for EPU. Waterford 3 plant data from Cycle 11 and Cycle 12 has been used in development of the CENTS model for EPU conditions.

[1] Louisiana Power and Light (i.e., Entergy) letter W3P85-3218 to the USNRC dated October 10, 1985 as amended by letter W3P85-0048 to the USNRC dated March 14, 1986.

The following transients have recently been experienced at Waterford 3 and were utilized in the development of the Waterford 3 long term cooling (LTC) model for EPU.

- Turbine trip from 100% power - February 14, 2003.
  The turbine trip had the steam bypass control system (SBCS) available to mitigate the transient. A reactor power cutback signal was automatically generated and quickly lowered reactor power to within the capability of the SBCS.
- Feedwater pump trip from 100% power - June 3, 2001.
  This caused a reactor power cutback to be initiated. The control systems operated as designed and there were no challenges to any of the safety systems.
- Reactor trip from approximately 82% power - February 13, 2001.
  A component failure caused turbine governor valve #3 to cycle open and closed, which caused an increase in power, and reactor trip on variable over power trip (VOPT). The plant operated as designed with feedwater and SBCS in automatic with steam generator pressure and level responding normally.

Industry experience at Arkansas Nuclear One Unit 2 during their 7.5% EPU demonstrated plant performance was adequately predicted under EPU conditions.

A scram, or the potential for a scram, from a high power level results in an unnecessary and undesirable plant transient cycle on the primary system, and the risk associated with the intentional introduction of a transient initiator, while small, should not be incurred unnecessarily.

The effects of the proposed EPU power level have been analytically addressed on a plant specific basis for Waterford 3. Transient analyses for Final Safety Analysis Report (FSAR) Chapter 15 events, such as loss of flow or turbine trip, have been performed to demonstrate that all safety criteria are met. No safety related systems will be significantly modified for EPU implementation. The minor changes in selected instrument setpoints do not merit the performance of large transient tests to confirm the acceptability of EPU operations at Waterford 3.

The Waterford 3 response to a turbine trip is described in Section 15.2.1.2 of the Waterford 3 FSAR and Section 2.13.2.1.2 of the EPU Report. Waterford 3 response to a loss of offsite power is described in Section 15.2.1.4 of the Waterford 3 FSAR and Section 2.13.2.1.4 of the EPU Report. Waterford 3 response to a loss of flow event is described in Section 15.3.2.1 of the Waterford 3 FSAR and in Section 2.13.3.2.1 of the EPU Report.

Transient mitigation capability is demonstrated through equipment surveillance tests required by Technical Specifications. Startup Physics tests are performed at the start of each fuel cycle. In addition, reload licensing analyses and physics tests at the start of each fuel cycle are performed to ensure that FSAR Chapter 15 safety analyses are maintained and all safety criteria remain met for each reload fuel cycle.

Waterford 3 will operate with a lower steam generator pressure and higher steam flow at hot full power for EPU. Control system adjustments will be made to compensate for these effects. These changes were considered and evaluated by the LTC computer code analyses which were performed to investigate and support the proposed control system adjustments for EPU. Large transient tests are not required to verify that these adjustments will provide adequate control system response since the LTC computer code has been sufficiently benchmarked with empirical data.

Waterford 3 Cycle 11 and Cycle 12 data has been used to tune the CENTS code. CENTS and LTC have been benchmarked against actual plant data that was gathered at plant power levels that are relatively closer to 3716 MWt power than the power level used to benchmark the original CESEC computer code for the startup Turbine Trip test discussed below. The power level used to benchmark the CESEC code was 16% lower than startup full power. The CENTS and LTC codes have been benchmarked against a power level that is 8% lower than 3716 MWt.

As documented in the November 13, 2003, letter submitting the EPU license amendment request, Waterford 3 will adopt the CENTS code, vice CESEC, for the performance of transient analyses as part of EPU. The CENTS code has been adopted as the primary analysis methodology for non-LOCA Chapter 15 transients on other Combustion Engineering designs including San Onofre Nuclear Generating Station (SONGS)-2/3, Arkansas Nuclear One (ANO)-2, and Palo Verde Nuclear Generating Station (PVNGS)-1/2/3. As documented in the CENTS Topical Report CENPD-282-P-A, the NRC safety evaluation for the CENTS code concluded CENTS is "acceptable for referencing in licensing actions with respect to the calculation of non-LOCA transient behavior in PWRs" and that "the CENTS code is acceptable for performing reload licensing analyses." The safety evaluation conditions on the applicability of the CENTS code are discussed in Table 12, p.A-28, of Attachment 1 ("Safety Evaluation Report Compliance") of the EPU license amendment request.

Waterford 3 plans to conduct load change tests and modification acceptance tests for adjustments to control systems as part of its EPU power ascension test plans. There is no impact upon these plans, nor any special testing needs, associated with the conditions imposed by NRC acceptance of the CENTS code.

Prior to EPU implementation, the simulator will be upgraded to provide operator training at EPU conditions.

Waterford 3 is designed and analyzed to respond safely in response to large transients. Unlike during startup tests, at-power plant behavior in response to transients has been well established over 19 years of commercial plant operation. Therefore, the analyses performed in support of EPU have increased credibility, due to greater benchmarking and greater plant knowledge, than those which were performed during the original startup test era. Thus, since acceptable plant response to such transients were demonstrated during original startup tests and since the changes associated with EPU have been extensively analyzed, it is unnecessary to expose the Waterford 3 plant to a potential plant transient associated with a large transient test. The ability of the Waterford 3 plant to properly respond to transients can be demonstrated through smaller scale transients, such as changes in load, which will demonstrate the adequacy of control system adjustments associated with EPU.

Detailed discussion of individual tests is provided below.

Remote Reactor Trip with Subsequent Remote Cooldown (SIT-TP-726)

The initial remote reactor trip with subsequent remote cooldown test concluded that the plant design was adequate to control the plant at remote locations. The test also concluded the operating procedures utilized during the performance of the remote trip and cooldown were adequate. The startup test for reactor trip and remote shutdown was performed from 20% power. There are no changes associated with EPU with impact on the ability of the reactor trip system or remote shutdown capability.

EPU will not modify the remote reactor trip circuit breakers (i.e., feeder breakers to the control element assemblies) nor any changes to operating procedures be required by EPU to perform a reactor trip. EPU will not affect transfer and isolation capability to the remote shutdown panel nor recommend any method changes to operating procedures for a remote plant shutdown and subsequent plant cooldown.

EPU will modify the remote shutdown panel with new atmospheric dump valves (ADV) controllers and rescaled plant instrument meters as required to reflect the new EPU operating parameters. These changes to the remote shutdown panel will be functionally tested during modification acceptance testing. These changes will not impact the manner in which control is transferred to the remote shutdown panel nor require an increased demand for remote shutdown capability following EPU.

The original test was performed from a relatively low power level, indicating there is no need to reperform the test at EPU conditions. The conclusions of this test remain valid for EPU.

<u>80% Total Loss of Flow Test/Natural Circulation Demonstration (SIT-TP-727)</u>

This test was performed to demonstrate the dynamic response of the Waterford 3 plant to a total loss of forced reactor coolant flow following sustained power operation is in accordance with design and that stable natural circulation can be established to maintain adequate core cooling.

- Plant response was recorded and compared to predictions generated with the CESEC code to verify the code for future transient analysis.
- Operations, startup test personnel, and vendor (Combustion Engineering) personnel performed an evaluation of the test results to verify that no design deficiencies or potential personnel safety hazards exist.
- Core Protection Calculator (CPC) generated trip response time was measured to determine the need for calculational uncertainty factors to be used in the core operating limit supervisory system (COLSS).
- The test demonstrated that natural circulation can be initiated.
- The test demonstrated that the natural circulation flow rate is adequate to maintain core cooling.

Additional startup tests to demonstrate natural circulation capability were conducted in SIT-TP-755, Natural Circulation Demonstration.

SIT-TP-727, 80% Total Loss of Flow Test/Natural Circulation Demonstration, was initiated by simultaneously tripping all four reactor coolant pumps. Once the transient was initiated the plant was allowed to respond without any operator action for 60 seconds. Plant response during this 60 second period was compared against that predicted by the CESEC code.

The natural circulation behavior of the Waterford 3 reactor coolant system (RCS) is essentially unchanged for EPU conditions. Unlike during startup tests, when plant data for such condition was not available in compiling analytical models for CESEC and other computer codes, Waterford 3 operational data has been used in the construction of the CENTS model which is being used for FSAR Chapter 15 safety analyses for EPU. Waterford 3 plant data from Cycle 11 and Cycle 12 has been used in development of the CENTS model for EPU conditions. Previous plant operating experience has also been factored into model development.

At hot standby natural circulation conditions, RCS decay heat and flow rate were measured to determine the power to flow ratio. The objective was to demonstrate adequate natural circulation flow by demonstrating that the (power/flow) ratio for natural circulation was less than that with the reactor at rated thermal power. This ensures the temperature rise across the core during natural circulation does not exceed the full power temperature rise. Note the temperature rise across the core is predicted to decrease with time, as decay heat levels decrease.

That test successfully demonstrated that the plant response was within design and that the plant was able to achieve natural circulation conditions following the transient. There has been no substantive change to the Waterford 3 RCS which would invalidate the conclusions of that startup test or its applicability to slightly higher power operation under EPU conditions. Note it was not necessary to perform the original startup test from maximum power conditions, thus there should be no need to perform this test for EPU conditions. Natural circulation behavior of the Waterford 3 RCS is essentially unchanged for EPU conditions. The flow coastdown analysis for EPU has been updated to account for increased steam generator tube plugging allowances and has been benchmarked against plant data. Thus, there is no reason to perform a similar test associated with EPU.

Loss of Offsite Power Trip (SIT-TP-728)

This test was performed to demonstrate plant performance under a total loss of AC power. This test verified that the reactor can be shutdown and hot standby conditions can be achieved and maintained using engineered safety features (ESF) power (4160 volt emergency diesel generator). Additionally, by simulating a total loss of onsite AC power, this test demonstrated the ability to remove decay heat with natural circulation flow in the reactor coolant system and with secondary feed from the steam-driven emergency feedwater pump.

The transient was initiated by tripping the main turbine from the control room, which subsequently resulted in a generator trip, causing the breakers to open and initiating a loss of offsite power. Unit startup transformers breakers had previously been opened, isolating alternate power to the station. The reactor was subsequently tripped from 20% power. Hot standby conditions were achieved and maintained for at least 30 minutes.

There are no modifications associated wtih EPU which would impact the ability of the emergency diesel generators (EDGs) to supply loads under loss of offsite power conditions. Technical specification response times for the EDG's are not being revised for EPU. The impact of EPU upon diesel loading has been assessed as reported in Section 2.3 of the EPU Report and found acceptable. The EPU will result in negligible changes in 4160 volt motor loads. The sequence of the safety related loads upon the EDGs is unchanged. Thus, there are negligible load changes to safety related buses and EDG's due to EPU. Thus, there are no changes that impact the ability to achieve reactor shutdown and hot standby conditions using ESF power.

Note that since Waterford 3 startup testing, the NRC has promulgated the Station Blackout Rule, 10CFR50.63. Entergy documented its response to the rule and associated Regulatory Guide 1.155 in memo W3P89-0510 dated April 14, 1989. As stated therein, plant procedures were reviewed and modified to meet the guidelines of NUMARC 87-00, and analyses were conducted to demonstrate the ability of Waterford 3 to cope with a four hour station blackout. These analyses were reviewed and, as required, updated to account for EPU operations.

The Waterford 3 station blackout evaluation is discussed in FSAR Section 8.1A. The impact of EPU upon Waterford 3's ability to cope with station blackout is discussed in Section 2.3.5 of the EPU Report. Section 2.3.5, of the EPU Report, concludes that the plant is capable of maintaining the RCS in hot standby condition and removing decay heat during the four hour station blackout post EPU.

<u>100% Turbine Trip (SIT-TP-740)</u>

This test was performed to demonstrate that the plant design was adequate to respond to a 100% power turbine trip and that plant systems responded in accordance with design. The data collected on plant response was used to verify the computer code predictions of CESEC, which was used for modeling plant transients. Dynamic response of main steam piping, monitored during the turbine trip, was also demonstrated to be acceptable.

The prerequisites for this test were that the reactor is stable with nuclear steam supply system control systems (SBCS, RRS, pressurizer level and pressure control) in automatic. The turbine would be manually tripped with no operator action taken for the first 60 seconds. The operators would then take actions per emergency operating procedures. Data would be collected and compared with the CESEC code predictions.

For startup testing, the test was satisfied by an unplanned transient. During power ascension, at approximately 88% power, a fire in the insulation of one of the main feed pumps occurred. The turbine was then manually tripped at approximately 84% power and the collected data was used for CESEC comparison.

For the NSSS control system setpoint evaluation for EPU, the LTC simulation code was used. The LTC computer code was benchmarked against Waterford 3 plant data:

- Cycle 12 steady state, taken on February 3, 2003
- Turbine trip at 100% power that occurred on February 14, 2003
- Feedwater pump trip at 100% power that occurred on June 3, 2001
- Reactor trip at approximately 82% power that occurred on February 13, 2001.

No major hardware modifications are planned for the Extended Power Uprate that will modify the nuclear steam supply system or main steam piping.

The CENTS computer code was used rather than the CESEC computer code for non-LOCA EPU safety analyses. Waterford 3 Cycle 11 and Cycle 12 data was used to tune the CENTS code for this purpose. CENTS and LTC have been benchmarked against actual plant data that was gathered at plant power levels that are relatively closer to 3716 MWt power than the power level used to benchmark the original CESEC computer code. The power level used to benchmark the CESEC code was 16% lower than startup full power. The CENTS and LTC codes have been benchmarked against a power level that is 8% lower than 3716 MWt. The dynamic response of main steam piping has been acceptable during all of these transients.

While Waterford 3 is designed and analyzed to respond safely to a turbine trip, introduction of a turbine trip is a transient initiator for the plant. Unlike during startup tests, at-power plant behavior in response to transients has been well established over 19 years of commercial plant operation. Thus, the analyses performed in support of EPU have increased credibility, due to greater benchmarking and greater plant knowledge, than those which were performed during the original startup test era. Thus, since acceptable plant response to a turbine trip was demonstrated during original startup tests and since the changes associated with EPU have been extensively analyzed, it is unnecessary to expose the Waterford 3 plant to the plant transient associated with a turbine trip test. The ability of the Waterford 3 plant to properly respond to transients can be demonstrated through smaller scale transients, such as changes in load, which will demonstrate the adequacy of setpoint changes to plant control systems associated with EPU.

Waterford 3 will operate with a lower steam generator pressure and higher steam flow at hot full power for EPU. SBCS adjustments will be made to compensate for these effects. With the increase in power level, the percent capacity of the SBCS has been reduced (physical capacity remains unchanged). Therefore, the reactor power cutback system will be put in service at 65% power (vice the current 70%) and the reactor trip on turbine trip setpoint will also be reduced to 65%. These effects were considered and evaluated by the LTC computer code analyses which were performed to investigate and support the proposed control system changes for EPU. A turbine trip at 100% power is not required to verify that these changes have been correctly implemented and, as previously stated, the LTC computer code has been sufficiently benchmarked with empirical data.

Previous operating experience indicates that the plant design is adequate to respond to a 100% power (~92% EPU power) turbine trip and that plant systems responded in accordance with design. Therefore there is no reason for exposing the plant to the risk involved in performing this test for EPU.

<u>RPCS Loss of Load/Feedwater Pump Testing (SIT-TP-749, SIT-TP-750, SIT-TP-751, SIT-TP-752, SIT-TP-753)</u>

The RPCS loss of load and loss of feedwater pump tests were designed to verify the interaction of the reactor power cutback and turbine systems in response to various transients such that there would be no engineered safety features actuation system (ESFAS) actuation, no reactor trip, and no lifting of primary or secondary code safeties.

This testing was deleted from the initial power ascension testing program. This system is a non-safety system with the purpose of avoiding reactor trips due to turbine trips, load rejections and loss of a feed pump. The primary purpose of this system is economic, to prevent a reactor trip; this system does not have a safety function. Should this system not stabilize the plant following a transient, safety systems will function to trip the plant.

As discussed in the 100% turbine trip section, the LTC code has been used determine system adjustments necessary to compensate for EPU parameter changes. This code has been benchmarked against various transients initiated between 82% and 100% power. The benchmarking results concluded that for the three transient cases considered, the agreement between the LTC code and the 100% loss of feedwater event was good, the agreement between the LTC code and the ~82% reactor trip was good and the agreement between the LTC code and the 100% power trip was good. Clearly empirical data and code refinement have enhanced the accuracy of the control system setpoint evaluation process. Based on this accurate modeling and actual plant transient response at current 100% power for both loss of load and loss of feed transients, and since the purpose of this system is economic, further transient testing is not required.

Natural Circulation Demonstration (SIT-TP-755)

The purpose of this test was to collect data to show that natural circulation flow conditions and heat removal capability are in accordance with design. The natural circulation conditions demonstrated during this test were:

a) initiation
b) steady state
c) reduced RCS pressure
d) isolated steam generator
e) recovery

Following the completion of the power-to-flow ratio determination of SIT-TP-727, steady state natural circulation conditions were maintained for approximately one hour.

Following the steady-state demonstration, all pressurizer heaters were secured, allowing RCS pressure to slowly decrease. Once the depressurization rate with no heaters energized had been determined, pressure was further reduced utilizing auxiliary spray. Natural circulation flow conditions were observed during this period at reduced pressure to demonstrate that natural circulation can be maintained at reduced system pressure and that proper loop subcooling could be maintained.

A demonstration of natural circulation with reduced heat removal capability was performed next. This test demonstrated that natural circulation can be maintained with one steam generator isolated. The secondary side of Steam Generator #2 was isolated by closing its atmospheric dump valve, main steam isolation valve, and securing feedwater to the steam generator. Natural circulation conditions were observed in this configuration and maintained for 30 minutes.

Natural circulation flow was shown to be able to adequately remove decay heat in each of the modes demonstrated in this test.

The natural circulation behavior of the Waterford 3 RCS is essentially unchanged for EPU conditions. Unlike during startup tests, when plant data for such condition was not available in compiling analytical models for CESEC and other computer codes, Waterford 3 operational data has been used in the construction of the CENTS model which is being used for FSAR Chapter 15 safety analyses for EPU. Waterford 3 plant data from Cycle 11 and Cycle 12 has been used in development of the CENTS model for EPU conditions. Previous plant operating experience has also been factored into model development.

Reactor coolant pump response times have previously been established. Plant surveillance procedures exist and are performed to ensure that plant equipment fulfills the Technical Requirements Manual Table 3.3-2 response time requirements assumed for CPC in safety analyses. The slightly higher decay heat of EPU will not impact the ability to initiate natural circulation or the ability of natural circulation to maintain adequate core cooling. The slight increase in decay heat associated with EPU would correspond to slightly earlier times and would have negligible impact on system capabilities; there is no change in thermal-hydraulic phenomena or behavior. There is no change in core mechanical design associated with EPU and only negligible changes in associated flow resistance. There will be no new system interactions related to the RCS thus EPU would not impact the natural circulation capability of Waterford 3, including under reduced RCS pressures or with an isolated steam generator. Therefore, this test need not be re-performed for EPU.