UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

March 5, 2007

Kevin T. Walsh, Vice President
of Operations
Waterford 3
Entergy Operations, Inc.
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - NRC BASELINE INSPECTION REPORT 05000382/2006012

Dear Mr. Walsh:

On February 12, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Waterford Steam Electric Station, Unit 3. The enclosed report documents the inspection findings, which were discussed on February 12, 2007, with you and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed plant and contract personnel who were involved in the performance of inservice inspection activities.

Based on the results of this inspection, the NRC has identified an issue that was evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that a violation is associated with this issue. This violation is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy. The noncited violation is described in the subject inspection report. If you contest the violation or significance of the noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Waterford Steam Electric Station, Unit 3 facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/ by GDR

William B. Jones, Chief
Engineering Branch 1
Division of Reactor Safety

Docket: 50-382
License: NPF-38

Enclosures:
Inspection Report 05000382/2006012
Attachments: A. Supplemental Information
B. Timeline of Events

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ROPreports
WAT Site Secretary **(AHY)**

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-382
License:	NPF-38
Report No.:	05000382/2006012
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	December 4, 2006, through February 12, 2007
Team Leader:	C. Paulk, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	G. George, Reactor Inspector, Engineering Branch 1 M. Holmberg, Reactor Inspector, Engineering Branch 1, Region III W. Sifre, Senior Reactor Inspector, Engineering Branch 1
Approved By:	William B. Jones, Chief, Engineering Branch 1

SUMMARY OF FINDINGS

IR 05000382/2006012; 12/4/2006 - 02/12/2007; Waterford Steam Electric Station, Unit 3; inservice inspection activities, Inspection Procedure 71111.08.

The report covered an announced inspection by regional inspectors. One Green noncited violation was identified. The significance of most findings is indicated by its color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self Revealing Findings

Cornerstone: Barrier Integrity

- Green. A noncited violation of Criterion XVI of Appendix B to 10 CFR Part 50 was identified for the failure to promptly identify and correct an adverse condition (i.e., steam generator batwing-to-wrapper bar welds not in accordance with design). Specifically, in May 2005, during Refueling Cycle 13, licensee personnel found that the batwing-to-wrapper bar welds were not in accordance with design drawings, but did not enter the adverse condition into the corrective action program until December 2006. This condition was entered into the corrective action program as Condition Report WF3-2006-04395.

  This finding was more than minor because by not promptly entering the non-conforming welds into the corrective action program and taking actions to correct the adverse condition, it became a more significant condition when two welds failed during Operating Cycle 14. Using the guidance of Appendix J to NRC Inspection Manual Chapter 0609, “Significance Determination Process,” the finding is determined to have very low safety significance (Green) because there was no tube degradation that exceeded 40 percent through-wall which did not increase in the large early release frequency. This finding had a crosscutting aspect in the area of problem identification and resolution (corrective action) program component. (Section 1R08.5).

B. Licensee-Identified Findings

None.

REPORT DETAILS

1. REACTOR SAFETY

1R08 Inservice Inspection Activities (71111.08P)

Inspection Procedure 71111.08P requires four samples, as identified in Sections 02.01, 02.02, 02.03, and 02.04.

.1 Examination Activities Other Than Steam Generator Tube Inspections, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control (Section 02.01)

a. Inspection Scope

The procedure requires the review of two to three types of nondestructive examination activities (NDE) (i.e. volumetric, surface, or visual). The inspectors reviewed the records of one visual examination, one magnetic particle (surface) examination, and two ultrasonic (volumetric) examinations. The ultrasonic examinations are listed below.

ISI REPORT NUMBER	SYSTEM/WELD AREA	EXAM METHOD
ISI-VT-06-064	Rigid Pressurizer System Restraint (RCCRR-0321)	Visual, VT-3
ISI-MT-06-001	Welded Support (43-WS-1) to 40-Inch Main Steam Pipe	Magnetic Particle
ISI-UT-05-029	Steam Generator 2 Main Steam Nozzle-to-Top Head-To-Dome Weld (04-030)	Ultrasonic
2002-238	Steam Generator 1 Hot leg Nozzle-to-Vessel Weld (03-010)	Ultrasonic
BOP-PT-06-011	SW-22 Final Weld Inspection	Penetrant

For each of the activities reviewed, the inspectors reviewed the examination records, NDE procedures, NDE staff certification records and interviewed the licensee inservice inspection program owner to determine if these activities performed were in accordance with site procedures and the applicable ASME Code requirements.

For the NDE examinations identified above, no recordable indications were identified or accepted for continued service and no Code recordable indications had been identified within the past two refueling outages. Therefore, the inspectors could not evaluate the licensee's resolution of Code recordable indications and this portion of the inservice inspection sample was considered unavailable for review.

In addition to NDE examinations, the procedure requires the inspectors review one to three welds performed on the pressure boundary of Class 1 systems. The inspectors reviewed welding to replace Safety Injection Valve SI-1421B and a drip pot upstream of Main Steam Isolation Valve 2.

The inspectors completed an additional review of welding activities to repair batwings (antivibration bars)-to-wrapper bar weld connections in Steam Generator 2. The inspectors reviewed the welder qualification mockup, welder qualification records, procedure qualification records, weld procedure specifications, acceptance criteria, and weld records. The inspectors performed these activities to verify that the weld connections met the design of Steam Generator 2.

The inspectors completed the one sample required by Section 02.01.

b. Findings

No findings of significance were identified.

.2 Pressurized-Water Reactor Vessel Upper Head Penetration Inspection Activities (Section 02.02)

a. Inspection Scope

The inspection requirements for this section parallel the inspection requirement steps in Section 02.01. The inspectors observed the nondestructive examinations on nine reactor vessel upper head penetrations. There were eight control element drive mechanism penetrations and one incore instrumentation penetration examinations observed.

The inspectors verified that the nondestructive activities were performed in accordance with the requirements of NRC Order EA-03-009. The nondestructive examinations performed during the NRC inspection did not reveal any defects or indications.

The inspectors completed the one sample required by Section 02.02.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Inspection Activities (PWR) (Section 02.03)

a. Inspection Scope

The inspectors performed a review of boric acid corrosion control procedures, interviewed licensee inspectors who were qualified to perform the boric acid inspections, performed a visual inspection of safety injection system valves located at the 35-foot elevation in containment, and observed a licensee's inspector perform a VT-2 visual examination of Valve SI-304B (2B safety injection tank drain valve) bolted connection. The inspectors performed these activities to verify that the licensee's visual examinations to detect leakage emphasized locations where boric acid leaks would cause degradation and to evaluate if boric acid leaks and/or component degradation were properly documented.

The inspectors reviewed four boric acid evaluations associated with minor boric acid deposits identified at the 2B reactor coolant pump seals, Steam Generator 2 cold leg piping, high pressure safety injection system flow element, and at assorted safety injection system valve packing and body-to-bonnet joints during the previous operating cycle. For these components, the inspectors evaluated the licensee's boric acid evaluations and associated corrective actions to determine if the ASME Code requirements associated with leakage or structural integrity were maintained (e.g., minimum wall thickness or bolt integrity).

The inspectors completed the one sample required by Section 02.03.

b. <u>Findings</u>

No findings of significance were identified.

.4 <u>Steam Generator Tube Inspection Activities (Section 02.04)</u>

a. <u>Inspection Scope</u>

Section 02.04 requires an assessment of insitu pressure testing screening criteria to assure consistency between assumed nondestructive examination flaw sizing accuracy and data from the Electric Power Research Institute (EPRI) examination technique specification sheets; a comparison of the predicted versus actual size and number of flaws detected; a confirmation that the scope of examinations and expansion criteria met the technical specification requirements, the EPRI Guidelines, and commitments made to the NRC; a verification that the licensee enveloped any new degradation mechanisms; a confirmation that all areas of potential degradation were inspected; a confirmation that any repairs were approved for use at the site; a confirmation that the plugging criteria were adhered to; an assessment of evaluation of any leakage >3 gallons per day; a confirmation that the probes and equipment were qualified; and, a review of corrective actions for any loose parts identified during the inspections.

The inspectors reviewed the in-situ pressure testing screening criteria to confirm that the criteria were in accordance with EPRI Guidelines, technical specifications, and commitments made to the NRC.

The inspectors reviewed licensee's Report ER-W3-2006-0180-000, "RF14 – Steam Generator Degradation Assessment," Revision 0. The purpose of the assessment is to identify degradation mechanisms and, for each mechanism, to determine proper detection technique, determine number of tubes, establish structural limits, and establish flaw growth rates.

The inspection procedure specified confirmation be made that the steam generator tube eddy-current testing scope and expansion criteria met technical specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors' review determined that the steam generator tube eddy-current testing scope and expansion criteria were being met.

The inspection procedure also specified that, if the licensee identified new degradation mechanisms, then verify that the licensee had fully enveloped the problem in an analysis and had taken appropriate corrective actions before plant startup. At the time of this inspection, licensee engineers were evaluating degradation mechanisms that were the result of batwing failures. Additional information is provided in Section 1R08.5 below.

The inspection procedure also required confirmation that all areas of potential degradation were being inspected, especially areas which were known to represent potential eddy-current testing challenges (e.g., top-of-tubesheet, tube support plates, and U-bends). The inspectors confirmed that all known areas of potential degradation, including eddy-current testing-challenged areas, were included in the scope of inspection and were being inspected.

The inspection procedure further required that repair processes being used were approved in the technical specifications for use at the site. The inspectors verified that the repair criteria were in accordance with EPRI Guidelines, technical specifications, and commitments made to the NRC.

The inspection procedure required confirmation that the technical specification plugging limit was being adhered to, and determination whether depth sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. The inspectors confirmed that the licensee was adhering to these specifications.

The inspection procedure stated that if steam generator leakage >3 gallons per day was identified during operations or during post-shutdown visual inspections of the tubesheet face, then assess whether the licensee had identified a reasonable cause and corrective actions for the leakage based on inspection results. The inspectors did not conduct any assessment because this condition did not exist.

The inspection procedure required confirmation that the eddy-current testing probes and equipment were qualified for the expected types of tube degradation and assessment of the site-specific qualification of one or more techniques. The inspectors observed portions of the eddy-current testing performed. During these examinations, the inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used; (2) probe position location verification was performed; (3) calibration requirements were adhered to; and (4) probe travel speed was in accordance with procedural requirements.

Finally, the inspection procedure specified the review of one to five samples of eddy-current testing data if questions arose regarding the adequacy of eddy-current testing data analyses. The inspectors did not identify any results where eddy-current testing data analyses adequacy was questionable.

The inspection performed met the requirements for the sample size of Section 02.04.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems (Section 02.05)

a. Inspection Scope

This portion of the inspection report details the inspectors' review of licensee personnel's followup activities to batwing failures in Steam Generator 2. As part of their review, the inspectors were to complete the following:

- Develop a detailed background and timeline for licensees'batwing failures (Attachment 2)

- Review the design requirements of the steam generators, particularly concerning batwing supports and structural integrity

- Evaluate the potential effects from the June 2005 power uprate

- Assess the licensee's problem identification and resolution efforts associated with the batwing failures

- Evaluate whether there were prior opportunities to have addressed and prevented the batwing failures

- Evaluate the potential for the issue to represent a generic concern with other facilities with similar batwing designs and/or configurations

- Monitor and assess selected repair activities and non-destructive examinations that the licensee or their contractors performed in conjunction with the batwing failures (Documented in 1R08.4 of this report)

- Determine and review the licensees'primary leak detection and loose parts monitoring capabilities

- Assess the licensees' contingencies and any planned limitations or action levels for plant startup and continued operation

b. Description of Event and Observations

Waterford 3 is a pressurized water reactor designed by Combustion Engineering Inc.. The Combustion Engineering, Inc. design incorporates two vertical shell, U-tube steam generators. The high temperature reactor coolant flows through the U-tube shaped steam generator tubes, which form the boundary between the primary and secondary sides of the steam generator. Cooling water flows upward through the vertical shell portion, cooling the steam generator tubes to generate steam. To prevent tube damage due to flow-induced vibration, each steam generator is equipped with 350 anti-vibration bars, or batwings. Each batwing is placed diagonally between the upper portions of adjacent steam generator tubes. Each batwing is connected to the tube bundle structure on the upper end by a double sided fillet weld to a wrapper bar surrounding the tube bundle and on the lower end by a batwing support structure in the center of the tube bundle.

On May 3, 2005, while performing an eddy-current inspection of the steam generator tubes during Refueling Outage 13, an analyst determined that two adjacent batwings had moved in Steam Generator 2. On May 10, 2005, a video inspection confirmed that the lower ends of two batwings had detached from the batwing support structure. This condition was documented in Condition Report WF3-2005-01977.

In addition, licensee personnel identified, from the video inspection, that a number of upper end batwing-to-wrapper bar welds were single sided fillet welds. These welds did not conform to manufacturing specifications, which called for double sided fillet welds. The upper end batwing-to-wrapper bar welds were determined to be intact and placed into service. Although the nonconforming welds were reviewed by a contractor, the nonconforming welds were not entered into the corrective action program and corrected by the licensee until prompted by the inspectors in December 2006.

As a corrective action for the detached lower ends of the batwings, licensee personnel implemented a protective strategy of plugging and stabilizing steam generator tubes in areas that could be affected by the broken batwings. Additionally, Operational Decision Making Instruction was revised to include conservative action levels for operators to mitigate a primary-to-secondary leakage event. This included a plan to visually inspect the batwings for more damage during Refueling Outage 14 (December 2006).

After the identification of initial batwing failures, licensee personnel generated an operational experience report, disseminating potential generic concerns. Similar plants of concern were St. Lucie 2, Palisades, and San Onofre Units 2 and 3. Batwing failures were not identified at any of the plants above. A search of operational experience databases indicated the batwing failures are unique to Steam Generator 2 at Waterford 3.

Starting with Operating Cycle 14, June 2005 to December 2006, the licensee operated in an approved uprated power condition. The Facility Operating License was amended to increase reactor core power level from 3441 MWt to 3716 MWt. Licensee personnel evaluated the potential effects of the power uprate causing additional batwing failures. The evaluation determined that there would be a low probability of additional failures specifically caused by the power uprate. The evaluation was based on engineering analysis and the knowledge that initial batwing failures occurred in pre-uprate power conditions.

On December 4, 2006, during Refueling Outage 14, licensee personnel identified additional lower end batwing failures in Steam Generator 2. On December 8, 2006, a video inspection identified two upper end batwings that had broken free from the single sided fillet weld attachments on the upper end. The video inspection also identified additional upper end batwing-to-wrapper bar welds, which did not conform with original design. These conditions were documented in Condition Reports WF3-2006-03966, WF3-2006-04122, WF3-2006-04138, WF3-2006-04144, WF3-2006-04395 and WF3-2006-04203.

As a corrective action for the additional batwing failures and nonconforming/broken batwing failures, licensee personnel continued the implementation of the protective strategy by plugging and stabilizing additional steam generator tubes in areas that could be affected by the broken batwings. Licensee personnel restored upper end batwing-to-wrapper bar welds in affected regions of the tube bundle to original design specifications. Additionally, licensee personnel revised the Operational Decision Making Instruction to include even more conservative action levels for operators to mitigate a primary to secondary leakage event. Furthermore, licensee personnel installed additional loose parts monitors to indicate impending batwing damage. As a prerequisite for continued operation past Refueling Outage 14, licensee personnel documented their commitments to implement the aforementioned actions in Letter W3F1-2006-0070.

c. Findings

Introduction. The inspectors identified a noncited violation of very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion XVI for the licensees' failure to promptly identify and correct an adverse condition involving nonconforming welds at the upper and batwing-to-wrapper bar.

Description. The licensee identified, in May 2005, that a number of upper end batwing-to-wrapper bar welds did not conform to manufacturing specifications. However, licensee personnel did not enter the nonconforming condition into the corrective action program until being prompted by the inspectors in December 2006. Corrective actions were not taken and two of the welds failed during Operating Cycle 14.

Analysis. The inspectors identified the failure to promptly identify and correct the nonconforming welds in May 2005 to be a performance deficiency. As a result of not taking corrective action to repair the batwing-to-wrapper bar connections, two welds failed during Operating Cycle 14. The inspectors found this to be more than minor because the lack of corrective actions resulted in a more significant condition (i.e., failed welds).

The significance of this finding was assessed using the guidance of Appendix J, to NRC Inspection Manual Chapter 0609, “Significance Determination Process,” as having very low safety significance (Green). This was based on no tube degradation that exceeded 40 percent through-wall, and therefore, there was not an increase in the large early release frequency.

The inspectors determined that the finding had a crosscutting aspect in the area of problem identification and resolution (corrective action program component). The failure to promptly identify the condition in their corrective action program caused the licensee to miss an opportunity to thoroughly evaluate and resolve the condition before the two batwing-to-wrapper bar single sided fillet weld failures occurred during Operating Cycle 14.

Enforcement. In accordance with 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” the licensee are required to promptly identify and evaluate conditions adverse to quality. Contrary to this, in May 2005, licensee personnel failed to enter the adverse condition into their corrective action program and correct nonconforming welds.

This item has been determined to be of very low safety significance and has been entered into the licensee’s corrective action program as Condition Report WF3-2006-04395. This is being treated as a noncited violation, consistent with Section VI.A if the Enforcement Policy: NCV 05000382/2006012-001, Failure to Promptly Identify and Correct an Adverse Condition (Welds Not In Accordance With Design).

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

On February 12, 2007, the inspectors presented the inspection results to Mr. K. Walsh, Vice President, Operations, and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was provided and examined during this inspection. However, no proprietary information is included in this report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

K. Cook, Acting General Plant Manager, Operations
T. Gaudet, Manager, Quality Assurance
J. Holman, Manager, Nuclear Engineering
B. Lanka, Manager, Design Engineering
R. O'Quinn, Senior Engineer, Steam Generator Inservice Inspection
B. Proctor, Manager, System Engineering
R. Putnam, Manager, Engineering
J. Ridgel, Director Nuclear Safety Assurance
B. Williams, Director, Engineering
R. Williams, Engineer, Licensing

NRC personnel

K. Karwoski, Senior Level Advisor for Steam Generators and Material Inspections, Office of Nuclear Reactor Regulation
G. Larkin, Senior Resident Inspector
E. Murphy, Senior Materials Engineer, Office of Nuclear Reactor Regulation
D. Overland, Resident Inspector

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000382/2006012-001	NCV	Failure to Promptly Identify and Correct an Adverse Condition (Welds Not In Accordance With Design) (Section 1R08.5)

DOCUMENTS REVIEWED

Boric Acid Corrosion Control

NUMBER	TITLE	REVISION / DATE
05-255	SI MV-303B Boric Acid Residue at Bolted Connection	June 8, 2005

Boric Acid Corrosion Control

NUMBER	TITLE	REVISION / DATE
05-256	RCP-2B Pump Cover has Dry Boric Acid Deposits	June 8, 2005
05-259	Suction Elbow off 2A Steam Generator Cold Leg has Boric Acid Staining	June 9, 2005
06-0431	HPSI 2B Flow Element White Boric Acid Underneath Flow Element	May 9, 2006
1F-7530-2	Byron Jackson Reactor Coolant Pump Drawing	E
CEP-NDE-0402, Attacment 9.1	ASME Section XI Leakage Test VT-2 - Regenerative Heat Exchanger Cubicle	April 1, 2002
NOECP-107	Boric Acid Corrosion Control Program	1
UNT-006-031	Identification and Evaluation of Boric Acid Leakage	1
UNT-007-027	Control of Boric Acid Corrosion on the Reactor Coolant System	5

Condition Reports

CR-WF3-2005-01622
CR-WF3-2005-01861
CR-WF3-2005-01977
CR-WF3-2005-02749
CR-WF3-2005-02751
CR-WF3-2005-03217
CR-WF3-2005-03710
CR-WF3-2005-04064
CR-WF3-2005-04179
CR-WF3-2006-01086
CR-WF3-2006-01197
CR-WF3-2006-01256
CR-WF3-2006-03966
CR-WF3-2006-04088
CR-WF3-2006-04104
CR-WF3-2006-04122
CR-WF3-2006-04138
CR-WF3-2006-04144
CR-WF3-2006-04191
CR-WF3-2006-04203
CR-WF3-2006-04395
CR-WF3-2007-00304

Drawings

NUMBER	TITLE	REVISION
1C83486	Steam Generator Batwing Weld Repair Option 1	0

Drawings

NUMBER	TITLE	REVISION
1C83488	Steam Generator Batwing Weld Repair Option 3	0

Engineering Requests

NUMBER	TITLE	REVISION
ER-W3-2005-0433-000	Leak on line 2MS2-47B, drip pot upstream of Main Steam Isolation Valve #2	0
ER-W3-2005-0446-000	Replace MS drip pot piping with stainless steel	0
ER-W3-2006-0339-000	Steam Generator Batwing Failures	0
ER-W3-2006-0362-000	Enhance the Steam Generator Batwing Support/Wrap-Around-Bar Welds	0
ER-W3-2006-0375-000	Install Loose Parts Monitoring Sensors on the Steam Generators	0
ER-W3-2006-0375-001	Installation of Temporary Loose Parts Monitoring Equipment for SG 1 and SG 2	0

Inservice Inspection Examinations

NUMBER	TITLE/DESCRIPTION	DATE
2002-238	Steam Generator No. 1 Hot leg Nozzle-to-Vessel Weld (03-010)	April 1, 2002
ISI-MT-06-001	Welded Support (43-WS-1) to 40 Inch Main Steam Pipe	November 29, 2006
ISI-UT-05-029	Steam Generator No. 2 Main Steam Nozzle-to-Top Head-to Dome Weld (04-030)	May 30, 2005
ISI-VT-06-064	Rigid Pressurizer System Restraint (RCCRR-0321)	November 29, 2006

Inservice Inspection Personnel Certification Records

Level	Method	Number of Staff Reviewed
II	Magnetic Particle Examination	1
II	Visual VT-3	2
II	Ultrasonic Examination	2

Miscellaneous

NUMBER	TITLE	REVISION / DATE
	Nondestructive Examination Procedure (PT) Qualification	May 22, 2001
	Nondestructive Examination Procedure (PT) Qualification	August 24, 2004
	QAD/QAI ANI/ANII Demonstration Record (PT)	January 29, 1990
ER-W3-2006-0180-000	RF14 – Steam Generator Degradation Assessment	0
LO-WLO-2006-0048	W3 SG Management Program Focused Assessment	February 6-9, 2006
LTR-CDME-07-9	Transmittal of Waterford SG32 R67 C99 Wear Discussion	
W3F1-2006-0024	12-Month Special Report SR-06-001-00 on the 13th Refueling Steam Generator Tube Inservice Inspection Waterford Steam Electric Station, Unit 3	May 17, 2006
W3F1-2006-0070	Entergy Actions to Address RF14 Batwing Failures Waterford Steam Electric Station, Unit 3 Docket No. 50-382	

Procedures

NUMBER	TITLE	REVISION
9.5	NDT Personnel Certification and Qualification Procedure for ANSI/ASNT CP 189 Compliance	5
A-08	Certification of N.D.E. Tech. Personnel	12
ANATEC-08	Certification of NDT Personnel (Eddy Current Method)	20
CEP-NDE-0955	Program Section for Reactor Vessel Head VT Examination (Alloy 600)	1A
CWTR3-400-001	Multifrequency Eddy Current Examination of Non-Ferromagnetic Steam Generator Tubing	03
EN-DC-317	Entergy Steam Generator Administrative Procedure	2
EN-MA-118	Foreign Material Exclusion	2
ER-W3-2006-0331-000	Steam Generator ECT Data Analysis for Waterford 3	0
INQ-2	Personnel Qualification and Certification Procedure	1
ML-QAP-9	Certification of NDE Personnel (ET)	8
NOECP-252	Steam Generator Inservice Eddy Current Testing	10
NOECP-257	Steam Generator Secondary Side Inspections	4
PQ & C-01	Written Practice for Personnel Qualification and Certification in Electromagnetic Testing	5
QAP-0101	Eddy Current Testing Personnel Qualification and Certification Procedure	17
QAP-18.02	Training, Qualification, and Certification of NDT Personnel	4
SEP-A600-001	Alloy 600 Management Program	0
TYG-100	Personnel Qualification Procedure	2

Procedures

NUMBER	TITLE	REVISION
WEC 9.2	Qualification, Training and Certification of Nondestructive Testing Personnel	4

Procedure Qualification Record

NUMBER	TITLE/DESCRIPTION	REVISION / DATE
PQR 107	GTAW & SMAW of P No. 8 material with NPS 6 Sch 40 pipe thickness	1
PQR 170	GTAW & SMAW of P No. 8 material with 1.5 in. plate thickness	1
PQR 591	GMAW of P No. 1 material with 0.375 in plate thickness	February 21, 1997

Welding Procedures

NUMBER	TITLE	REVISION
MRS-SSP-2065-CWTR3	Waterford Nuclear Generating Station Unit 3: Batwing Welding Procedure	0
PS-S-100W	Preassigned In-process Inspections	0

Welding Procedure Specifications

NUMBER	TITLE	REVISION
1F43; Ar SA-GMAW	Welding Procedure Specification	0
E-P8-T-A8, Ar	Welding Procedure Specification	0

Work Order

86597-01

Timeline of Events

January 1985 - Tube wear due to batwing vibration first discovered at San Onofre Nuclear Generating Station during eddy current inspection of Unit 2 steam generators. Several plants susceptible to the problem were Waterford 3, Palo Verde, Palisades, and St. Lucie 2.

August 1985 - To mitigate the potential for a tube leak caused by batwing vibration, Waterford begins a “plugging and staking” campaign of 149 steam generator tubes adjacent to the central cavity, as recommended by Combustion Engineering. Completed during RFO 4.

October 18, 2003 - Waterford 3 shuts down for RFO 12. Batwings are intact based on eddy current signals. A chemical cleaning is conducted, which includes sludge lancing and a high volume upper bundle flush.

November 17, 2003 - Waterford 3 starts up from RFO 12.

April 17, 2005 - Waterford 3 commences RFO 13.

April 24, 2005 - Eddy current testing commenced.

April 25, 2005 - 0400 Eddy current engineer log entry noting larger than expected number of batwing wear and appearing to have significant growth of 20-25%.

May 2, 2005 - Investigation indicates batwing movement from RFO 12 location.

May 3, 2005 - Analysts determine that there must be two adjacent batwing failures to explain signals.

May 3, 2005 - CR-WF3-2005-01977 initiated

May 10, 2005 - Video inspection of the SG #2 upper batwing to wrap-around-bar welds is completed. All welds intact but SG #2 welds noted to be single sided.

May 11, 2005 - Video inspection of the SG #2 lower batwings completed confirming two failed batwings.

May 12, 2005 - Video inspection of SG #1 upper batwing welds is completed. No failures noted. Welds noted to be two sided.

May 15, 2005 - Video inspection of SG #1 lower batwing completed. No failures noted.

November 23, 2006 - WF3 shuts down to commence RFO 14.

December 4, 2006 - Visual inspections identify additional degradation of batwings in SG #2.

December 8, 2006 - 0354 Visual inspections identify batwings 84/85 is broken free at the top attachment weld and is “missing.”

December 8, 2006 - Visual inspections identify batwing 108/109 is broken free at the top attachment weld.

December 12, 2006 - Visual inspections identify batwing attachment welds on SG #2 are in noncompliance with the design requirements.

December 14, 2006 - Entergy implements a comprehensive tube plugging and stabilization plan that provides a defense-in-depth configuration to protect the integrity of the active tubes of the steam generators.

December 18, 2006 - Entergy repairs deficient upper batwing to wrap-around-bar welds, in the stay cavity region, to provide a structural connection which will meet required load conditions.

December 20, 2006 - Waterford sends Letter W3F1-2006-0070 to the NRC to commit to actions to address additional batwing failures and Steam Generator Tube Rupture mitigation strategies. Strategies include addition of loose parts monitor and more conservative action limits for primary to secondary leakage.