UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

August 11, 2011

Joseph Kowalewski, Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-0751

Subject: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – NRC INTEGRATED INSPECTION REPORT 05000382/2011003

Dear Mr. Kowalewski:

On June 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Waterford Steam Electric Station, Unit 3 facility. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 8, 2011, with you and other members of your staff.

The inspections 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 personnel.

The report documents two NRC-identified findings and two self-revealing findings which were evaluated under the risk significance determination process as being of very low safety significance (Green). Three of these findings were determined to involve a violation of NRC requirements. Additionally, two licensee-identified violations which were also determined to be of very low safety significance are listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations consistent with Section 2.3.2.a of the NRC Enforcement Policy.

If you contest any of the noncited violations, 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, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 East Lamar Blvd., Suite 400, Arlington, Texas 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at the facility. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC inspectors at the facility

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you choose to provide one for cases where a response is not required, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary, information so that it can be made available to the Public without redaction.

Sincerely,

/RA/

James Drake, Chief (Temporary)
Project Branch E
Division of Reactor Projects

Docket No.: 50-382
License No.: NPF-38

Enclosure: NRC Inspection Report 05000382/2011003
w/Attachment: Supplemental Information

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SUNSI Rev Compl.	☑ Yes ☐ No	ADAMS	☑ Yes ☐ No	Reviewer Initials	RVA
Publicly Avail	☑ Yes ☐ No	Sensitive	☐ Yes ☑ No	Sens. Type Initials	RVA

SRI:DRP/E	RI:DRP/E	SPE:DRP/E	C:DRS/EB1	C:DRS/EB2
MDavis	DOverland	RAzua	TRFarnholtz	NFO'Keefe
/RA/via E	***/JDrake for/***	***/RA/***	***/RA/***	***/JMateychick for/***
8/1/11	8/11/11	8/10/11	8/4/11	8/8/11
C:DRS/OB	C:DRS/PSB1	C:DRS/PSB2	C:DRS/TSB	C:DRP/E
MSHaire	MHay	GEWerner	DPowers	JDrake
/BLarson for/	***/RA/***	***/RA/***	***/RA/***	***/RA/***
8/4/11	8/8/11	8/4/11	8/5/11	8/11/11

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket No.:	50-382
License No.:	NPF-38
Report:	05000382/2011003
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Killona, LA
Dates:	April 1 through June 30, 2011
Inspectors:	M. Davis, Senior Resident Inspector D. Overland, Resident Inspector C. Smith, Project Engineer R. Azua, Senior Project Engineer S. Makor, Reactor Inspector M. Williams, Reactor Inspector L. Carson, II, Senior Health Physicist N. Greene, Health Physicist P. Elkmann, Senior Emergency Preparedness Inspector
Approved By:	James Drake, Acting Branch Chief Project Branch E Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000382/2011003; 04/01/2011–06/30/2011; Waterford Steam Electric Station, Unit 3, Integrated Resident and Regional Report; Maintenance Effectiveness, Plant Modifications, Post-Maintenance Testing, and Problem Identification and Resolution

The report covered a 3-month period of inspection by resident inspectors and announced baseline inspections by regional based inspectors. Two NRC-identified and two self-revealing findings were identified with three being noncited violations. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." The cross-cutting aspect was determined using Inspection Manual Chapter 0310, "Components within the Cross Cutting Areas." Findings for which the significance determination process 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 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

- Severity Level IV. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.71(e) because the licensee did not revise the Final Safety Analysis Report (FSAR), as updated, with information consistent with plant conditions. Specifically, the licensee did not update Section 5.4.1.3 of the FSAR for Waterford Steam Electric Station, Unit 3, following modifications to the reactor coolant pump vapor seals in 2007 and 2009, respectively. As a result, the licensee did not promptly identify and correct FSAR noncompliance. The licensee entered this issue into their corrective action program for resolution as CR-WF3-2010-7421. The planned corrective actions include revising the FSAR, as updated, and replacing the degraded reactor coolant pump seals during the next two refueling outages.

  The inspectors considered this issue to be within the traditional enforcement process because it has the potential to impede or impact the NRC's ability to perform its regulatory function. The inspectors used the NRC Enforcement Policy to evaluate the significance of this violation. The inspectors concluded that the violation was more than minor because the longstanding and incorrect information in the FSAR, as updated, had a material impact on safety and licensed activities. The material impact was that the modifications to the reactor coolant pump vapor seals, created the conditions for a reactor coolant pump seal loss of coolant accident inside containment, which could have potentially impacted licensed activities. The inspectors determined the violation was a Severity Level IV (very low safety significance) since the information that was not updated in the FSAR, was not used to make an unacceptable change to the facility nor did it impact a licensing or safety decision by the NRC. The inspectors determined there was a cross-cutting aspect in the corrective action component of the problem identification and resolution area. Specifically, the licensee did not thoroughly evaluate and take adequate actions in a

timely manner to update the FSAR to be consistent with plant conditions (P.1(c)) (Section 1R18).

- Green. A self-revealing finding occurred because maintenance technicians did not follow written procedures during the calibration of a level switch that controls feedwater heater drain valve FHD703A. Specifically, the technicians did not perform concurrent verification checks as required by documented work order instructions (WO-00180716) and procedures to ensure that personnel restore and/or manipulate components to the correct position following maintenance. As a result, the feedwater heater drain valve was left in a closed position, which caused a spurious isolation of a string of feedwater heaters. The isolation of the feedwater heaters caused operators to down power the reactor to approximately 72 percent. The licensee entered this issue into their corrective action program for resolution as CR-WF3-2009-7420. The immediate corrective actions included restoring the feedwater heater drain valve to its proper position.

  The finding was more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the human error caused an event that upset plant stability during power operation. The inspectors evaluated this finding using Inspection Manual Chapter 0609 Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The inspectors determined that the finding was of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. The finding had a crosscutting aspect in the work practices component of the human performance area because the licensee's personnel proceeded in the face of uncertainty or unexpected circumstances (H.4(a)) (Section 4OA2.3).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR 50.65(a)(3) because the licensee did not evaluate or adequately monitor activities associated with the condition of the condensate and refueling water storage pool structures. Specifically, the licensee did not evaluate the internal condition of the storage pools through the performance of appropriate preventive maintenance activities and did not evaluate these activities at least every refueling cycle, where practical, per industry-wide operating experience. As a result, there was no preventive maintenance developed for this activity even though industry-wide operating experience documented previous issues of concrete deterioration due to contact with boric acid over a long period of time. The licensee entered this issue into their corrective action program for resolution as CR-WF3-2011-1168. The planned corrective actions include the development of appropriate preventive maintenance activities to examine the internal condition of the storage pool structures during refuel outages.

The finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, no preventive maintenance to monitor the internal condition of the storage pools, would impact the reliability of the structures. The inspectors evaluated this finding using Inspection Manual Chapter 0609 Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The inspectors determined that the finding was of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of a safety function of a system or a single train for greater than its technical specification completion time, and did not screen potentially risk significant due to external events. The finding had a crosscutting aspect in the operating experience component of the problem identification and resolution area because the licensee did not implement and institutionalize operating experience through changes to station processes, procedures, equipment, and training programs (P.2(b)) (Section 1R12).

- <u>Green</u>. A self-revealing noncited violation of Technical Specification 6.8.1.a occurred because the licensee did not implement written procedures and work order instructions. Specifically, maintenance personnel did not follow Procedure ME-007-005, "Time Delay Relay Setting Check, Adjustment, and Functional Test", during the lifting leads process for restoration of a time delay relay (EG EREL2327-C) associated with an 'A' emergency diesel generator (EDG) maintenance activity. As a result, the 'A' EDG output breaker did not automatically close during technical specification surveillance testing because the leads on the relay were wired incorrectly. The licensee entered this issue into their corrective action program for resolution as CR-WF3-2011-3190. The immediate corrective action included the re-wiring of the relay to allow the 'A' EDG to automatically close to the safety-related bus.

  The finding was more than minor because it was associated with the human and equipment performance attributes of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the licensee did not ensure the availability, reliability and capability of the 'A' EDG through the use of human error prevention techniques. The inspectors performed the initial significance determination for the diesel generator output breaker failure. The inspectors used the NRC Inspection Manual Chapter 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The finding screened to a Phase 2 significance determination because it involved a potential loss of one train of safety related equipment for longer than the technical specification allowed outage time. A Region IV senior reactor analyst performed a Phase 2 significance determination and used the pre-solved worksheet from the "Risk Informed Inspection Notebook for the Waterford-3 Nuclear Power Plant," Revision 2.01a. The senior reactor analyst considered the output breaker a part of the emergency diesel generator component boundary. Assuming a one year exposure period, the finding was potentially Yellow,

which warranted further review. Therefore, the senior reactor analyst performed a bounding Phase 3 significance determination. The analyst determined that the finding was of very low safety significance (Green). The bounding change to the core damage frequency was approximately 5.4E-7/year. The dominant core damage sequences included loss of offsite power events, failure of the output breaker recovery action, independent failure of the other emergency diesel generator and failure to recover offsite power in 4 hours. Equipment that helped mitigate the risk included the ability of an operator to recover the output breaker. The finding had a crosscutting aspect in the work practices component of the human performance area because the licensee did not communicate human performance error prevention techniques, such as self and peer checking, and proper documentation of activities (H.4(a)) (Section 1R19).

B. Licensee-Identified Violations

Two violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and their associated corrective action tracking numbers are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

The Waterford Steam Electric Station, Unit 3, began the inspection period at approximately 85 percent power. On April 5, 2011, Operators commenced a down power to conduct activities associated with refueling outage 17. On May 14, Operators started to increase power to 100 percent. The Unit remained at 100 percent power for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity and Emergency Preparedness

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- On April 12, 2011, train 'B' of the shutdown cooling system during maintenance of the 'A' train

- On April 21, 2011, train 'A' of the high pressure safety injection system following an extended system outage

- On April 22, 2011, train 'B' of the low pressure safety injection system following an extended system outage

- On June 6, 2011, emergency diesel generator B while emergency diesel generator A was inoperable during corrective maintenance

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, updated final safety analysis report, technical specification requirements, administrative technical specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also inspected accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were

no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four (4) partial system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

.2 System Walkdowns associated with Temporary Instruction (TI) 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems."

a. Inspection Scope

The inspectors reviewed isometric drawings and piping and instrumentation diagrams, and conducted walkdowns to verify that the licensee had drawings that described the subject system configurations. Specifically, the inspectors verified the following related to the isometric drawings for the high and low pressure safety injection systems:

- High point vents were identified
- High points that did not have vents were acceptably recognizable
- Other areas where gas could accumulate and potentially impact subject system operability, such as at orifices in horizontal pipes, isolated branch lines, heat exchangers, improperly sloped piping, and under closed valves, were acceptably described in the drawings or in referenced documentation
- Horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified
- All pipes and fittings were clearly shown.

The inspectors verified that the drawings were up-to-date with respect to recent hardware changes, and that any discrepancies between the as-built configurations and the drawings were documented and entered into the licensee's corrective action program for resolution. Documents reviewed are listed in the attachment to this report.

This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later inspection report.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors conducted fire protection walkdowns that were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- On April 11, 2011, the reactor containment building (RCB) fire area, all elevations
- On April 14, 2011, the fuel handling building (FHB) fire area, all elevations
- On April 19, 2011, the reactor auxiliary building (RAB) fire area, fire zone RAB 7A, relay room ‘A'
- On April 22, 2011, the reactor auxiliary building (RAB) fire area, fire zone RAB 7B, relay room ‘B'

The inspectors reviewed areas to assess if licensee personnel had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features, in accordance with the licensee’s fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant’s Individual Plant Examination of External Events with later additional insights, their potential to affect equipment that could initiate or mitigate a plant transient, or their impact on the plant’s ability to respond to a security event. Using the documents listed in the attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee’s corrective action program. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four (4) quarterly fire-protection inspection samples as defined in Inspection Procedure 71111.05-05.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the updated final safety analysis report, the flooding analysis, and plant procedures to assess susceptibilities involving internal flooding; reviewed the corrective action program to determine if licensee personnel identified and corrected flooding problems; and verified that operator actions for coping with flooding can reasonably achieve the desired outcomes. The inspectors also inspected the areas listed below to verify the adequacy of equipment seals located below the flood line, floor and wall penetration seals, watertight door seals, common drain lines and sumps, sump pumps, level alarms, and control circuits, and temporary or removable flood barriers. Specific documents reviewed during this inspection are listed in the attachment.

- On May 10, 2011, wet cooling tower areas

These activities constitute completion of one (1) flood protection measures inspection sample as defined in Inspection Procedure 71111.06-05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Inspection Activities Other Than Steam Generator Tube Inspection, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, and Boric Acid Corrosion Control (71111.08-02.01)

a. Inspection Scope

The inspectors observed three nondestructive examination activities and reviewed eight nondestructive examination activities that included two types of examinations. The licensee did not identify any relevant indications accepted for continued service during the nondestructive examinations.

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Pressurizer Surge Nozzle Inner Radius (05-014)	Ultrasonic Testing
Reactor Coolant System	Pressurizer Nozzle to Bottom Head Weld (05-009)	Ultrasonic Testing
Reactor Coolant System	Nozzle to Safe-End Circumferential Weld (15-006)	Ultrasonic Testing

The inspectors reviewed records for the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Nozzle to Safe-End Circumferential Weld (26-001)	Phased Array Ultrasonic Testing
Reactor Coolant System	Nozzle to Safe-End Circumferential Weld (26-006)	Phased Array Ultrasonic Testing
Reactor Coolant System	Nozzle to Safe-End Circumferential Weld (26-010)	Phased Array Ultrasonic Testing
Reactor Coolant System	Nozzle to Safe-End Circumferential Weld (25-029)	Phased Array Ultrasonic Testing
Reactor Coolant System	Nozzle to Safe-End Circumferential Weld (16-017)	Phased Array Ultrasonic Testing
Reactor Coolant System	Nozzle to Safe-End Circumferential Weld (06-006)	Phased Array Ultrasonic Testing
Reactor Coolant System	Nozzle to Safe-End Circumferential Weld (15-008)	Phased Array Ultrasonic Testing
Reactor Coolant System	Nozzle to Safe-End Circumferential Weld (15-009)	Phased Array Ultrasonic Testing

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with the ASME Code requirements and applicable procedures. The inspectors also verified the qualifications of all nondestructive examination technicians performing the inspections were current.

The inspectors reviewed 4 welds on the reactor coolant system pressure boundary.

The inspectors reviewed records for the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD TYPE
Safety Injection (SI-405B)	EC-14767 FW 9A	Fillet
Safety Injection (SI-405B)	EC-14767 FW 2	Fillet
Safety Injection (SI-405B)	EC-14767 SW 16	Fillet
Safety Injection (SI-405B)	EC-14767 FW 1	Fillet

The inspectors verified, by review, that the welding procedure specifications and the welders had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also verified, through observation and record review, that essential variables for the welding process were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.01.

b. Findings

No findings were identified.

.2 Vessel Upper Head Penetration Inspection Activities (71111.08-02.02)

a. Inspection Scope

The inspectors reviewed the results of licensee personnel's visual inspection of pressure-retaining components above the reactor pressure vessel head to verify that there was no evidence of leaks or boron deposits on the surface of the reactor pressure vessel head or related insulation. The inspectors verified that the personnel performing the visual inspection were certified as Level II and Level III VT-2 examiners. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.02.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control Inspection Activities (71111.08-02.03)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown as specified in Procedure NOECP-107, "Boric Acid Corrosion Control Program (BACCP)," Revision 3. The inspectors also reviewed the visual records of the components and equipment. The inspectors verified that the visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components. The inspectors also verified that the engineering evaluations for those components where boric acid was identified gave assurance that the ASME Code wall thickness limits were properly maintained. The inspectors confirmed that the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME Code. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.03.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities (71111.08-02.04)

a. Inspection Scope

The inspectors assessed the in-situ 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. The inspectors assessed the appropriateness of tubes selected for in-situ pressure testing, observed in-situ pressure testing, and reviewed the in-situ pressure test results. Specific documents reviewed during this inspection are listed in the attachment.

In addition, the inspectors reviewed both the licensee site-validated and qualified acquisition and analysis technique sheets used during this refueling outage and the qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy, tubing, equipment, technique, and analysis had been identified and qualified through demonstration. The inspectors reviewed acquisition technique and analysis technique data sheets.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess the licensee's prediction capability. The inspectors compared the previous outage operational assessment predictions with the flaws identified during the current steam generator tube inspection effort. The number of identified indications fell below the range of prediction but was consistent with historical predictions.

The inspection procedure specified confirmation that the steam generator tube eddy current test scope and expansion criteria meet technical specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors compared the recommended test scope to the actual test scope and found that the licensee had accounted for all known flaws and had, as a minimum, established a test scope that met technical specification requirements, EPRI guidelines, and commitments made to the NRC. The scope of the licensee's eddy current examinations of tubes in both steam generators included:

- 100 percent bobbin examination full length of tubing
- 100 percent hot leg top of tube sheet
- 100 percent Rows 1 and 2 u-bend rotating pancake coil
- 100 percent dented tube supports at egg crates greater than 2 Volts
- 20 percent dented diagonal bar and vertical strap greater than 2 Volts
- 20 percent free span dings greater than 5 Volts

- Cold leg top of tube sheet periphery exam for loose parts

The inspection procedure specified that, if new degradation mechanisms were identified, the licensee would verify the analysis fully enveloped the problem of the extended conditions including operating concerns and that appropriate corrective actions were taken before plant startup. No new degradation mechanisms were identified.

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

The inspection procedure further required verification that repair processes being used were approved in the technical specifications. The inspectors confirmed that the repair processes being used were consistent with the technical specifications requirements.

The inspection procedure also required confirmation of adherence to the technical specification plugging limit, unless alternate repair criteria have been approved. The inspection procedure further requires 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 determined that the technical specification plugging limits were being adhered to (i.e., 40 percent maximum through-wall indication).

If steam generator leakage greater than 3 gallons per day was identified during operations or during post shutdown visual inspections of the tubesheet face, the inspection procedure required verification that the licensee had identified a reasonable cause based on inspection results and that corrective actions were taken or planned to address the cause for the leakage. The inspectors did not conduct any assessment because this condition did not exist.

The inspection procedure required confirmation that the eddy current test probes and equipment were qualified for the expected types of tube degradation and an assessment of the site-specific qualification of one or more techniques. The inspectors observed portions of the eddy current tests. 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; and (4) probe travel speed was in accordance with procedural requirements. The inspectors performed a review of site-specific qualifications of the techniques being used.

These actions constitute completion of the requirements of Section 02.04.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems (71111.08-02.05)

a. Inspection scope

The inspectors reviewed 30 condition reports which dealt with inservice inspection activities and found the corrective actions for inservice inspection issues were appropriate. The specific condition reports reviewed are listed in the documents reviewed section. From this review the inspectors concluded that the licensee has an appropriate threshold for entering inservice inspection issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry inservice inspection operating experience. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements of Section 02.05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On June 21, 2011, the inspectors observed a crew of licensed operators in the plant's simulator to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems and training was being conducted in accordance with Entergy Operations, Inc. procedures. The inspectors evaluated the following areas:

- Licensed operator performance
- Crew's clarity and formality of communications
- Crew's ability to take timely actions in the conservative direction
- Crew's prioritization, interpretation, and verification of annunciator alarms
- Crew's correct use and implementation of abnormal and emergency procedures
- Control board manipulations
- Oversight and direction from supervisors
- Crew's ability to identify and implement appropriate technical specification actions and emergency plan actions and notifications

The inspectors compared the crew’s performance in these areas to pre-established operator action expectations and successful critical task completion requirements. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) quarterly licensed-operator requalification program sample as defined in Inspection Procedure 71111.11-05. Specific documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk significant systems:

- On May 17, 2011, condensate and refueling water storage pool structures
- On May 18 , 2011, the licensee’s maintenance rule (a)(3) periodic evaluation

The inspectors reviewed events such as where ineffective equipment maintenance has resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65(b)
- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or (a)(2)
- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1)

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two (2) quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

Introduction. The inspectors identified a Green finding associated with a noncited violation of 10 CFR 50.65(a)(3) because the licensee did not evaluate or adequately monitor activities associated with the internal condition of the condensate and refueling water storage pool structures. Specifically, the licensee did not evaluate the internal condition of the storage pools through the performance of appropriate preventive maintenance activities and did not evaluate these activities at least every refueling cycle, where practical per industry-wide operating experience.

Description. During a review of an issue of concern related to the condensate storage pool (CSP), the inspectors questioned the adequacy of the maintenance rule monitoring of the CSP and refueling water storage pool (RWSP) structures. Specifically, the inspectors wondered why the licensee did not develop preventive maintenance activities and take into account industry-wide operating experience during the periodic evaluation of these structures. Both the CSP and RWSP are safety-related seismic Category I structures located within the Reactor Auxiliary Building (RAB). The design of both the CSP and RWSP are similar, since both are: (1) located within a walled-in room of the RAB; (2) lined with stainless steel on the floor and the walls; and (3) both have unlined concrete walls and ceiling. The RWSP contains borated water with a boron concentration of between 2050 and 2900 parts per million (ppm) with a pH of approximately 4.5. The licensee monitors the outside surfaces of both the CSP and the RWSP and monitors the tell tale drain system for the RWSP structure. However, the licensee informed the inspectors that the interiors of these structures had not been accessed or inspected since original construction. The licensee only monitors the outside surfaces of both structures because they considered the storage pools inaccessible. The inspectors reviewed the drawing associated with these storage pools and noted that both structures have an accessible concrete floor plug on the +21 foot level elevation of the RAB. The inspectors determined that it was feasible to inspect both storage pools by removing the accessible concrete floor plugs. The inspectors' concern was that over the years the concrete portions of the structures could degrade and challenge the reliability of the storage pools, which was similar to issues identified at Salem, Indian Point, and Seabrook. These issues involved water collecting between the stainless steel liner and concrete walls that caused degradation of the surrounding concrete.

Additionally, the inspectors reviewed the drawings of the RWSP to look for possible ways that the concrete portion of the RWSP could be degraded. The inspectors noted

that a 6 inch pump recirculation line enters both pools at the +16.5 foot mean sea level (MSL) elevation. When the licensee uses these pump recirculation lines, the humidity of the air in these pools would increase, and humid environments can adversely affect the concrete. Furthermore, the boric acid in the RWSP could splash onto the concrete portions of the structure and could be detrimental to the concrete, as noted in Information Notice 2004-05, “Spent Fuel Pool Leakage to Onsite Groundwater.”

The inspectors also noted that the licensee’s maintenance rule Procedure, EN-DC-150, “Condition Monitoring of Maintenance Rule Structures,” states in part, that Entergy Sites are committed to NUREG 1522, “Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures.” Section 5.9 of NUREG 1522 describes the approach for inaccessible areas, and notes in part that:

> *Thus, areas inaccessible for periodic inspections, such as underground or underwater portions of the structures, need to be realistically evaluated for susceptibility to degradation mechanisms and sustained as well as infrequent stressors. These evaluations should include consideration of site-specific (plant-specific) characteristics, experience at other nuclear power plants, and the history or testing of such features (e.g., piles) under similar conditions. Where feasible, these features should be closely inspected at an appropriate interval (e.g., 10 years) using divers or ground exploration. The periodic inspection (evaluation) of inaccessible areas should be considered for incorporation in all requirement and guideline documents for the maintenance of structures at nuclear power plants.*

The inspectors determined that the licensee did not inspect the inside of the storage pools because the licensee considered the pools inaccessible. However, this was contrary to their condition monitoring of maintenance rule structure procedure. The inspectors also noted that an industry standard to inspect concrete was ACI 349.3R, “Evaluation of Existing Nuclear Safety-Related Concrete Structures.” This standard would require inspection of the interior surfaces of both the CSP and RWSP every 5 years. The inspectors concluded that the licensee did not evaluate the internal condition of the storage pools through the performance of appropriate preventive maintenance activities and did not evaluate these activities at least every refueling cycle, where practical, per industry-wide operating experience. The inspectors noted that the licensee had opportunities to evaluate appropriate preventive maintenance activities as a part of their two year review and through various operating experience. The licensee entered this issue into their corrective action program for resolution as CR-WF3-2011-1168. The planned corrective actions included the development of appropriate preventive maintenance activities to examine the internal of the storage pool structures during the refuel outages.

<u>Analysis.</u> The performance deficiency was that licensee did not evaluate or adequately monitor activities associated with the condition of the condensate and refueling water storage pool structures. Specifically, the licensee did not evaluate the internal condition of the storage pools through the performance of appropriate preventive maintenance activities and did not evaluate these activities at least every refueling cycle, where

practical, per industry-wide operating experience. The inspectors determined that it was reasonable for the licensee to be able to foresee and prevent occurrence of this deficiency. The finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, no preventive maintenance to monitor the internal condition of the storage pools would impact the reliability of the structures. The inspectors evaluated this finding using Inspection Manual Chapter 0609 Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The inspectors determined that the finding was of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of a safety function of a system or a single train greater than its technical specification completion time, and did not screen potentially risk significant due to external events. The finding had a cross-cutting aspect in the operating experience component of the problem identification and resolution area because the licensee did not implement and institutionalize operating experience through changes to station processes, procedures, equipment, and training programs (P.2(b)).

Enforcement. Title 10 of CFR 50.65 (a)(1)specifies, in part, that licensee's shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience." Title 10 of CFR 50.65 (a)(3) states, in part, that "performance and condition monitoring activities and associated goals and preventive maintenance activities shall be evaluated at least every refueling cycle provided the interval between evaluations does not exceed 24 months. The evaluations shall take into account, where practical, industry-wide operating experience." Contrary to the above, as of March 3, 2011, the licensee had not taken into account industry wide operating experience, in that accessible areas of the Condensate Storage Pool and the Refueling Water Storage Pool had not been inspected and effectively controlled through the performance of appropriate preventive maintenance. However, because this finding was of very low safety significance and it was entered into the corrective action program as CR-WF3-2011-1168, this violation was treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000382/2011003-01: Failure to evaluate and adequately monitor activities associated with the internal conditions of the condensate and refueling water storage pool structures).

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed licensee personnel's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- On April 4, 2011, scheduled and emergent maintenance activities on the ‘A’ emergency diesel generator with impending adverse weather in the area of the site

- On April 14, 2011, scheduled maintenance activities during shutdown operation while the ‘B’ emergency diesel generator and train B of the low pressure safety injection pump were out of service with deviations in containment closure

- On May 14, 2011, emergent maintenance activities on the ‘AB’ emergency feedwater pump with scheduled maintenance on the boric acid makeup pump

- On June 6, 2011, scheduled corrective maintenance outage for the ‘A’ emergency diesel generator due to this important mitigation equipment being out of service

- On June 23, 2011, emergent corrective maintenance on train A of the essential chiller system with the ‘B’ emergency diesel generator air receiver being out of service

The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of five (5) maintenance risk assessments and emergent work control inspection samples as defined in Inspection Procedure 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- On April 4, 2011, operability evaluation of the ‘A’ emergency diesel generator

- On May 14, 2011, operability evaluation of the ‘AB’ emergency feedwater pump

- On June 13, 2011, operability evaluation of the 'B' emergency diesel generator following alarms indicating an over-speed trip had occurred while the engine was being shut down

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that technical specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and updated final safety analysis report to the licensee personnel's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three (3) operability evaluations inspection samples as defined in Inspection Procedure 71111.15-04

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Permanent Modifications

a. Inspection Scope

The inspectors reviewed key affected parameters associated with materials, replacement components, equipment protection from hazards, operations, flow paths, pressure boundary, structural, process medium properties, licensing basis, and failure modes for the permanent modifications listed below.

- On April 1, 2011, engineering change EC-6256, added a valve connection in the reactor coolant pump seal drain headers

- On April 1, 2011, engineering change EC-18520, added a tee connection in the vapor stage drain lines for all four reactor coolant pumps

The inspectors verified that modification preparation, staging, and implementation did not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; post modification testing will maintain

the plant in a safe configuration during testing by verifying that unintended system interactions will not occur; systems, structures and components' performance characteristics still meet the design basis; the modification design assumptions were appropriate; the modification test acceptance criteria will be met; and licensee personnel identified and implemented appropriate corrective actions associated with permanent plant modifications. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two (2) samples for permanent plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

Introduction. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.71(e) because the licensee did not revise the final safety analysis report (FSAR) as updated with information consistent with plant conditions. Specifically, the licensee did not update Section 5.4.1.3 of the FSAR for Waterford Steam Electric Station, Unit 3 following modifications to the reactor coolant pump vapor seals in 2007 and 2009, respectively.

Description. The inspectors identified that the final safety analysis report (FSAR) as updated was inconsistent with current plant conditions during a review of an issue of concern. The inspectors noted that section 5.4.1.3 of the FSAR stated, in part, that coolant entering the seal chambers was cooled and collected in a closed system so that the reactor coolant leakage to containment was essentially zero. However, during Refueling Outage 15 (March 2007) and Refueling Outage 16 (October 2009), the licensee implemented Engineering Changes EC-6256 and EC-18520, respectively. These modifications redirected the reactor coolant pump vapor seal leakage flow to floor drains inside containment and collected the leakage in the containment sump as unidentified leakage. Based on the inspector's review of these modifications, the inspectors determined that the plant conditions changed but the licensee did not review and update the FSAR. The inspectors concluded that the licensee missed opportunities to review and update the FSAR through their engineering change process. As a result, the licensee did not promptly identify and correct FSAR noncompliance. The licensee entered this issue into their corrective action program for resolution as CR-WF3-2010-7421. The planned corrective actions included the licensee revising the FSAR as updated and replacing the degraded reactor coolant pump seals during the next two refueling outages.

Analysis. This issue was a performance deficiency because the licensee had reasonable opportunity to identify and correct the FSAR to be consistent with current plant conditions. The inspectors considered this issue to be within the traditional enforcement process because it had the potential to impede or impact the NRC's ability to perform its regulatory function. The inspectors used the NRC Enforcement Policy to evaluate the significance of this violation. The inspectors concluded that the violation was more than minor because the longstanding and incorrect information in the FSAR as updated had a material impact on safety and licensed activities. The material impact was that the modifications created a small loss of coolant accident scenario in

containment that could have potentially impacted licensed activities. The inspectors determined that the violation was a Severity Level IV (very low safety significance) since the erroneous information in the FSAR, not updated, was not used to make an unacceptable change to the facility nor did it impact a licensing or safety decision by the NRC. The inspectors determined that there was a crosscutting aspect associated with this finding in the corrective action component of the problem identification and resolution area. Specifically, the licensee did not thoroughly evaluate and take adequate actions in a timely manner to update the FSAR to be consistent with plant conditions (P.1(c)).

Enforcement. Title 10 of CFR Part 50.71(e) states, in part, that each person licensed to operate a nuclear power reactor under the provisions of § 50.22, shall update periodically the FSAR, originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee or prepared by the applicant or licensee pursuant to Commission requirement since the submittal of the original FSAR, or as appropriate, the last update to the FSAR under this section. Contrary to the above, as of November 10, 2010, the licensee had not updated Section 5.4.1.3 of the Waterford Steam Electric Station, Unit 3 FSAR following modifications to the reactor coolant pump vapor seals in 2007 and 2009, respectively. As a result, the licensee did not promptly identify and correct FSAR noncompliance. However, because this finding was a Severity Level IV and it was entered into the corrective action program as CR-WF3-2010-7421, this violation was being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000382/2011003-02: Failure to Update Section 5.4.1.3 of the FSAR for Waterford Steam Electric Station, Unit 3 following modifications to the reactor coolant pump vapor seals).

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the post-maintenance test for the maintenance activities listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors evaluated these activities for the following (as applicable):

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed

- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the technical specification, the updated final safety analysis report, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the

inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether Entergy Operations, Inc. was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. The inspectors selected the following post-maintenance testing activities based on the ability of the structures, systems and components to affect risk:

- On April 16, 2011, scheduled maintenance activities on the ‘A’ high pressure safety injection pump

- On April 19, 2011, emergent corrective maintenance on the ‘B’ emergency diesel generator potential transformer

- On April 20, 2011, scheduled corrective maintenance on the ‘A’ station battery cell number one

- On April 21, 2011, emergent corrective maintenance on the ‘AB’ station battery cell number three

- On April 22, 2011, emergent corrective maintenance on shutdown cooling inside containment isolation valve (SI-405B)

- On April 23, 2011, emergent corrective maintenance on the ‘MA’ static uninterruptible power supply

- On April 30, 2011, emergent corrective maintenance on the A’ emergency diesel generator

- On June 7, 2011, scheduled corrective maintenance on the ‘A’ emergency diesel generator

These activities constitute completion of eight (8) post-maintenance testing inspection samples as defined in Inspection Procedure 71111.19-05. Specific documents reviewed during this inspection are listed in the attachment.

b. Findings

Introduction. The inspectors determined that a self-revealing Green finding associated with a noncited violation of Technical Specification 6.8.1.a occurred because the licensee did not implement written procedures and work order instructions. Specifically, the licensee did not follow Procedure ME-007-005, “Time Delay Relay Setting Check, Adjustment, and Functional Test”, during the lifting leads process for restoration of a time delay relay (EG EREL2327-C) associated with the train A emergency diesel generator (‘A’ EDG) output breaker.

Description. On April 30, 2011, during the performance of the train A integrated emergency diesel generator engineering safety features test, the diesel generator auto started, but did not energize safety busses 3A and 31A because the ‘A’ EDG output

breaker did not close as expected. As a result, Control Room Operators attempted to close the EDG output breaker manually but were unsuccessful. The licensee initiated a condition report to evaluate the cause of this condition. The licensee's investigation revealed that a time delay relay was not wired correctly. This time delay relay assures that another breaker opens prior to closing the 'A' EDG output breaker onto the safety-related bus. A review of historical documentation revealed that maintenance personnel performed preventive maintenance (PM) on this time delay relay during a scheduled six year maintenance overhaul of the diesel generator on November 2, 2010. The PM activity instructed the technicians to disconnect all electrical leads to the relay and remove them from the panel to perform a bench test in the electrical shop. Prior to disconnecting the leads, the technicians were required per ME-007-005, Attachment 12.2 to record the wire identification and termination on a lifted lead sheet to ensure the leads were properly returned to service. However, the maintenance personnel did not follow the procedure to restore the time delay relay to its proper terminal. No concurrent verification or approved design documentation was used to restore the lifted lead. The licensee also identified that the work order instructions did not contain a post maintenance test after re-installation of the relay. The inspectors noted that the post maintenance test for this particular relay can only be performed during an outage. The licensee performed the overhaul PM activity for the 'A' EDG on line and determined that the Lift Lead Verification Form constituted the post maintenance test for the time delay relay. The licensee entered this issue into their corrective action program for resolution as CR-WF3-2011-3190. The immediate corrective action included the re-wiring of the relay.

<u>Analysis.</u> The performance deficiency was that maintenance personnel did not implement written procedures and instructions. Specifically, maintenance personnel did not follow Procedure ME-007-005, Attachment 12.2 during the lifting leads process for restoration of a time delay relay associated with the output breaker. The inspectors determined that it was reasonable for the licensee to be able to foresee and prevent occurrence of this deficiency. The finding was more than minor because it was associated with the human and equipment performance attributes of the Mitigating Systems Cornerstone and affects the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the licensee did not ensure the availability, reliability and capability of the 'A' EDG through human error prevention techniques. The inspectors evaluated this finding using Inspection Manual Chapter 0609 Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The inspectors performed the initial significance determination for the diesel generator output breaker failure. The inspectors used the NRC IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The finding screened to a Phase 2 significance determination because it involved a potential loss of one train of safety related equipment for longer than the technical specification allowed outage time.

A Region IV senior reactor analyst performed a Phase 2 significance determination and used the pre-solved worksheet from the "Risk Informed Inspection Notebook for the Waterford-3 Nuclear Power Plant," Revision 2.01a. The output breaker was considered part of the emergency diesel generator component boundary. Assuming a one year exposure period, the finding was potentially Yellow, which warranted further review.

Therefore, the senior reactor analyst performed a bounding Phase 3 significance determination.

The analysts performed simplified calculations to determine the change to the core damage frequency (delta-CDF) for the diesel output breaker failure. During testing, the emergency diesel generator (EDG) started but the output breaker failed to close. During maintenance on or about November 2, 2010 craft personnel miss-wired a control-circuit relay. The miss-wired relay affected a breaker interlock such that the breaker would not automatically close to a dead bus. The licensee verified that the relay on train B was wired properly. The NRC inspectors reported that operators could not easily close the EDG output breaker from the control room (this was attempted following the breaker failure). The inspectors also stated that local breaker operation was possible at the breaker cubicle. In addition, operators were trained on local breaker operation and a procedure directed operators to close the breaker. In short, an operator would don electrical safety equipment and depress a button on the inside of the breaker panel. The NRC resident inspectors verified that this action would work.

The total exposure period spanned from November 2, 2010 to May 2, 2011 (182 days). This period included a refueling outage, which started on April 4. The analyst broke the exposure period up into the following segments.

- **At Power Time:** The at-power period was 153 days. For the next two days, all safety related equipment was available as the plant was cooled down for refueling preparations. The analyst grouped these two days with the at power days, which was conservative. The final 10 days of the exposure were at the back end of the outage. The reactor coolant system was filled and systems were returned to service. The analyst conservatively included these 10 days in with the at-power time. Therefore, the analyst used a total at-power exposure period of 165 days.
- **Mid-Loop:** The licensee conducted two mid-loop evolutions during the exposure period. The analyst calculated the delta-CDF for the mid-loop evolutions separately. Each mid-loop period was approximately two days. While the decay heat load was substantially lower during the second mid-loop, the analyst conservatively treated both mid-loop evolutions equally. The total mid-loop exposure period was 4 days.
- **Flood-Up Period:** For approximately 13 days, the reactor cavity was flooded to 23 feet above the reactor vessel flange to support refueling. Once the reactor vessel was flooded up, the core damage risk was substantially diminished. Therefore, the risk during this exposure period was considered negligible and was not quantified.

To evaluate the breaker malfunction, the analyst used the Waterford 3 SPAR Model, Revision 8.15 (Saphire 8), with a truncation limit of 1E-11. Breaker 3A-14 was not specifically identified in the SPAR. This breaker was normally considered to be within the EDG component boundary. Therefore, the analyst adjusted the EDG failure to start probability to account for the breaker malfunction. This adjustment included the subsequent recovery action (non-recovery probability). The equipment that helped mitigate the risk included the ability of an operator to recover the output breaker. The finding has a cross-cutting aspect in the work practices component of the human performance area because the licensee did not communicate human performance error

prevention techniques, such as self and peer checking, and proper documentation of activities (H.4(a)).

Enforcement. Technical Specification 6.8.1.a states, in part, that written procedures shall be established, implemented, and maintained for activities described in Appendix A of the Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. Specifically, Section 9 of RG 1.33, Appendix A states, Maintenance that can affect the performance of safety related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances. Contrary to the above, on November 2, 2010, maintenance personnel did not follow Procedure ME-007-005, "Time Delay Relay Setting Check, Adjustment, and Functional Test," Attachment 12.2 during the lifting leads process for restoration of a time delay relay (EG EREL2327-C) associated with the 'A' EDG output breaker. As a result, EDG A output breaker did not automatically close during technical specification surveillance testing on April 30, 2011. However, because this finding was of very low safety significance and it was entered into the corrective action program as CR-WF3-2011-3190, this violation was being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000382/2011003-04: Failure to implement written procedures for restoring a time delay relay associated with 'A' EDG output breaker)

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the Waterford Steam Electric Station, Unit 3 refueling outage, conducted on April 6, 2011 – May 11, 2011 to confirm that licensee personnel had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense in depth. During the refueling outage, the inspectors observed portions of the shutdown and cool-down processes and monitored licensee controls over the outage activities listed below.

- Configuration management, including maintenance of defense in depth, was commensurate with the outage safety plan for key safety functions and compliance with the applicable technical specifications when taking equipment out of service

- Clearance activities, including confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing

- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error

- Status and configuration of electrical systems to ensure that technical specifications and outage safety-plan requirements were met, and controls over switchyard activities

- Monitoring of decay heat removal processes, systems, and components
- Verification that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system
- Reactor water inventory controls, including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss
- Controls over activities that could affect reactivity
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the reactor containment building to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing
- Licensee identification and resolution of problems related to refueling outage activities

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) refueling outage inspection sample as defined in Inspection Procedure 71111.20-05.

b. Findings

Introduction. The inspectors identified an unresolved item (URI) associated with a loss of reactor coolant system inventory through incore instrumentation flanges, which occurred on April 30, 2011.

Description. On April 30, 2011, while the plant was in Mode 5 during Refueling Outage 17 (RFO-17), operators identified reactor coolant inventory coming from the incore instrumentation flanges. Based on the restoration from other maintenance activities, operators locked closed the pressurizer spray line vent valve RC-309 on the prior shift to return the valve to its normal position. However, the plant conditions at the time of restoration required RC-309 to be open to provide a reactor coolant vent path during the assembly of incore instrumentation (ICI) flanges. The inspectors noted that the loss of reactor coolant inventory occurred twice prior to the licensee securing from assembly of the ICI flanges. The inspectors questioned whether the operators understood the true condition of the plant since the closed vent path caused inaccurate reactor coolant system level indication. The inspectors also questioned the amount of reactor coolant inventory loss because operators identified water coming out of the ICI flanges twice while charging with the charging pumps. The inspectors reviewed the initial condition report and the maintenance work activities surrounding this evolution. However, the

licensee had not completed the apparent cause evaluation prior to the end of the inspection period.

This item was unresolved pending further review and investigation of the licensee's apparent cause evaluation such that the inspectors can determine if there are performance deficiencies associated with this loss of reactor coolant inventory event (URI 05000382/2011003-05, Loss of reactor coolant inventory during the assembly of incore instrumentation flanges)

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysts Report, technical specifications, and Entergy Operations, Inc. procedure requirements to ensure that the surveillance test activities listed below demonstrated that the SSCs being tested were capable of performing their intended safety functions. The inspectors either observed or reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning
- Evaluation of testing impact on the plant
- Acceptance criteria
- Test equipment
- Procedures
- Jumper/lifted lead controls
- Test data
- Testing frequency and method demonstrated technical specification operability
- Test equipment removal
- Restoration of plant systems
- Fulfillment of ASME Code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for returning tested systems, structures, and components not meeting the test acceptance criteria were correct
- Reference setting data

- Annunciators and alarms setpoints

The inspectors also verified that licensee personnel identified and implemented any needed corrective actions associated with the surveillance testing activity.

- On April 5, 2011, scheduled surveillance to perform simmer testing on the main steam safety valve

- On April 7, 2011, scheduled surveillance to verify operability of containment isolation actuation signal

- On April 12, 2011, emergent surveillance to verify operability of engineering safety features actuation signal lockout relays

- On April 19, 2011, scheduled surveillance to verify operability of the 'B' emergency diesel generator and engineering safety features

- On April 26, 2011, scheduled surveillance to perform local leak rate testing of penetration 41

- On May 9, 2011, scheduled surveillance to verify operability of the 'AB' emergency feedwater pump

- On June 7, 2011, scheduled surveillance to verify operability of the 'A' emergency diesel generator following corrective maintenance

These activities constitute completion of seven (7) surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05, including one (1) inservice test and one (1) containment isolation valve test. Specific documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector performed in-office reviews of Waterford Steam Electric Station Emergency Plan, Revision 40, and Procedure EP-001.001, "Recognition and Classification of Emergency Conditions," Revision 27. These revisions:

- Added the following emergency response organization positions: Emergency Operations Facility Manager, Emergency Operations Facility Communicator,

- Emergency Operations Facility Habitability Technician, Maintenance Coordinator (Technical Support Center), Operations Support (Operations Support Center), Information Coordinator (Joint Information Center), and Media Liaison (Joint Information Center);

- Deleted the following emergency response organization positions: Emergency Operations Facility Operations/Engineer Coordinator, Emergency Operations Facility Operations/Engineer Coordinator Assistant, Emergency Operations Facility Electrical, Emergency Operations Facility Mechanical, Emergency Operations Facility Instrument and Controls, Emergency Operations Facility Nuclear Engineering, Dose Assessment Coordinator (Technical Support Center), Dose Assessment Coordinator Assistant (Technical Support Center), Dose Assessor Communicator (Technical Support Center), Technical Support Center Lead Communicator, Technical Support Center Communicator, Governmental Affairs Coordinator (Joint Information Center), and Offsite Agency Coordinator (Joint Information Center);

- Defined that the Emergency Director (Control Room) transfers event command and control responsibilities to the Emergency Director (Emergency Operations Facility);

- Transferred oversight of Fire Brigade activities during an emergency from the Operations Support Center Manager to the Control Room Supervisor;

- Moved radiological assessment functions from the Technical Support Center to the Emergency Operations Facility;

- Moved engineering and event mitigation functions from the Emergency Operations Facility to the Technical Support Center;

- Defined that the Emergency Director (Emergency Operations Facility) and Emergency Plant Manager (Technical Support Center) each authorize emergency radiation exposures and the distribution of potassium iodide for emergency workers under their direction;

- Renamed the Emergency News Center to the Joint Information Center;

- Changed thirty-one emergency response organization titles;

- Clarified the staffing time goal for the Joint Information Center;

- Changed the provider of air ambulance services from West Jefferson Medical Center Air Care to Ochsner Flight Care; and

- Made minor editorial changes and corrections.

These revisions were compared to their previous revisions, to the criteria of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency

Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, to Nuclear Emergency Institute Report 99-01, "Methodology for Development of Emergency Action Levels," Revision 4, and to the standards in 10 CFR 50.47(b) to determine if the revisions adequately implemented the requirements of 10 CFR 50.54(q). These reviews were not documented in safety-evaluation reports and did not constitute approval of licensee-generated changes; therefore, these revisions are subject to future inspection. The specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two samples as defined in Inspection Procedure 71114.04-05.

b. Findings

No findings were identified.

Cornerstone: Occupational and Public Radiation Safety

2RS01 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

This area was inspected to: (1) review and assess licensee's performance in assessing the radiological hazards in the workplace associated with licensed activities and the implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures, (2) verify the licensee was properly identifying and reporting Occupational Radiation Safety Cornerstone performance indicators, and (3) identify those performance deficiencies that were reportable as a performance indicator and which may have represented a substantial potential for overexposure of the worker.

The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed walkdowns of various portions of the plant, performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation reported by the licensee in the Occupational Radiation Safety Cornerstone
- The hazard assessment program, including a review of the license's evaluations of changes in plant operations and radiological surveys to detect dose rates, airborne radioactivity, and surface contamination levels

- Instructions and notices to workers, including labeling or marking containers of radioactive material, radiation work permits, actions for electronic dosimeter alarms, and changes to radiological conditions

- Programs and processes for control of sealed sources and release of potentially contaminated material from the radiologically controlled area, including survey performance, instrument sensitivity, release criteria, procedural guidance, and sealed source accountability

- Radiological hazards control and work coverage, including the adequacy of surveys, radiation protection job coverage, and contamination controls; the use of electronic dosimeters in high noise areas; dosimetry placement; airborne radioactivity monitoring; controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools; and posting and physical controls for high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

- Audits, self-assessments, and corrective action documents related to radiological hazard assessment and exposure controls since the last inspection

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of the one (1) required sample as defined in Inspection Procedure 71124.01-05.

b. Findings

No findings were identified.

2RS02 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

This area was inspected to assess performance with respect to maintaining occupational individual and collective radiation exposures as low as was reasonably achievable (ALARA). The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee’s procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspectors interviewed licensee personnel and reviewed the following items:

- Site-specific ALARA procedures and collective exposure history, including the current 3-year rolling average, site-specific trends in collective exposures, and source-term measurements

- ALARA work activity evaluations/post job reviews, exposure estimates, and exposure mitigation requirements

- The methodology for estimating work activity exposures, the intended dose outcome, the accuracy of dose rate and man-hour estimates, and intended versus actual work activity doses and the reasons for any inconsistencies

- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

- Audits, self-assessments, and corrective action documents related to ALARA planning and controls since the last inspection

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of the one required sample as defined in Inspection Procedure 71124.02-05.

b. Findings

No findings were identified.

2RS03 In-plant Airborne Radioactivity Control and Mitigation (71124.03)

a. Inspection Scope

This area was inspected to verify in-plant airborne concentrations are being controlled consistent with ALARA principles and the use of respiratory protection devices on-site do not pose an undue risk to the wearer. The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee’s procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspectors interviewed licensee personnel, performed walkdowns of various portions of the plant, and reviewed the following items:

- The licensee’s use, when applicable, of ventilation systems as part of its engineering controls

- The licensee’s respiratory protection program for use, storage, maintenance, and quality assurance of NIOSH certified equipment, qualification and training of personnel, and user performance

- The licensee’s capability for refilling and transporting SCBA air bottles to and from the control room and operations support center during emergency

conditions, status of SCBA staged and ready for use in the plant and associated surveillance records, and personnel qualification and training

- Audits, self-assessments, and corrective action documents related to in-plant airborne radioactivity control and mitigation since the last inspection

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of the one (1) sample as defined in Inspection Procedure 71124.03-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the performance indicator data submitted by the licensee for the first quarter of 2011 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 0608, “Performance Indicator Program.”

This review was performed as part of the inspectors’ normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings were identified.

.2 Unplanned Scrams per 7000 Critical Hours (IE01)

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours performance indicator during the previous four quarter for the period from April 2010 through March 2011. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, was used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, event reports and NRC Inspection reports for the period of January 2010 through March 2011 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been

identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one (1) unplanned scrams per 7000 critical hours sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.3 Unplanned Scrams with Complications (IE02)

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications performance indicator during the previous four quarter for the period from April 2010 through March 2011. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, was used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection reports for the period of January 2010 through March 2011 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one (1) unplanned scrams with complications sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.4 Unplanned Power Changes per 7000 Critical Hours (IE03)

a. Inspection Scope

The inspectors sampled licensee submittals for the unplanned power changes per 7000 critical hours performance indicator during the previous four quarter for the period from April 2010 through March 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection reports for the period of January 2010 through March 2011 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been

identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one (1) unplanned transients per 7000 critical hours sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.5 Mitigating Systems Performance Index - Emergency ac Power System (MS06)

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index - emergency ac power system performance indicator for the period from the second quarter 2010 through the first quarter 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, mitigating systems performance index derivation reports, issue reports, event reports, and NRC integrated inspection reports for the period of January 2010 through March 2011 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one (1) mitigating systems performance index - emergency AC power system sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.6 Mitigating Systems Performance Index - High Pressure Injection Systems (MS07)

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index - high pressure injection systems performance indicator for the period from the second quarter 2010 through the first quarter 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance

Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, issue reports, mitigating systems performance index derivation reports, event reports, and NRC integrated inspection reports for the period of January 2010 through March 2011 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one (1) mitigating systems performance index - high pressure injection system sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.7 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors reviewed performance indicator data for the first through fourth quarters of 2010. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

The inspectors reviewed corrective action program records associated with high radiation area (greater than 1 rem/hr) and very high radiation area non-conformances. The inspectors reviewed radiological, controlled area exit transactions greater than 100 mrem. The inspectors also conducted walkdowns of high radiation areas greater than 1 rem/hr and very high radiation area entrances to determine the adequacy of the controls of these areas.

These activities constitute completion of the occupational exposure control effectiveness sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.8 Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences (PR01)

a. Inspection Scope

The inspectors reviewed performance indicator data for the second quarter 2010 through the first quarter 2011. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

The inspectors reviewed the licensee's corrective action program records and selected individual annual or special reports to identify potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose.

These activities constitute completion of the radiological effluent technical specifications/offsite dose calculation manual radiological effluent occurrences sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

.1 Routine Reviews of Identification and Resolution of Problems

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. The inspectors reviewed attributes that included the complete and accurate identification of the problem; the timely correction, commensurate with the safety significance; the evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews; and the classification, prioritization, focus, and timeliness of corrective actions. Minor issues entered into the licensee's corrective action program because of the inspectors' observations are included in the attached list of documents reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure, they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused their review on repetitive equipment issues, but also considered the results of daily corrective action item screening discussed in Section 4OA2.2, above, licensee trending efforts, and licensee human performance results. The inspectors nominally considered the 6-month period of January through June of 2011, although some examples expanded beyond those dates where the scope of the trend warranted.

The inspectors also included issues documented outside the normal corrective action program in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's corrective action program trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

These activities constitute completion of one (1) sample of a semi-annual trend inspection.

.3 Selected Issue Follow-up Inspection

a. Inspection Scope

The inspectors performed an in-depth review of the licensee's evaluation and corrective actions related to an unexpected isolation for a string of low pressure feedwater heaters. The inspectors reviewed the appropriateness of the assigned significance, the scope and depth of the causal analysis, and the timeliness of resolution. The inspectors assessed whether the evaluation identified likely causes for the issues and identified appropriate corrective actions to address the identified causes. The inspectors also conducted a review of the corrective actions to verify that appropriate measures were in place to prevent reoccurrence of the issue. In addition, the inspectors assessed whether the licensee's evaluation considered extent of condition, generic implications, common

cause, and previous occurrences. The inspectors reviewed the potential impact on nuclear safety and risk to verify that the licensee had taken corrective actions commensurate with the significance of the issue. The inspectors evaluated these actions against the requirements of the licensee's corrective actions program and performance attributes contained in IP 71152, Section 03.06.

These activities constitute completion of one (1) in-depth problem identification and resolution sample as defined in Inspection Procedure 71152-05.

b. Findings

Introduction. A Green, self-revealing finding occurred because technicians did not follow written procedures and instructions during the calibration of a level switch that controls feedwater heater drain valve FHD703A. Specifically, technicians did not perform concurrent verification checks as required by the work order instructions (WO-00180716) to ensure that personnel restored manipulated components to the correct position following maintenance.

Description. On December 6, 2009, following the power ascension to 100 percent from refueling outage 16, the control room received feedwater heater 5A HI/HI level alarm. Operations personnel took action to adjust the level set-point of the 5A feedwater heater to clear the alarm condition and approximately 30 minutes later isolated extraction steam to the 4A feedwater heater. On December 7, 2009, during an attempt to restore the 4A feedwater heater, the control room received another feedwater heater 5A HI/HI level alarm. With the normal drain valve for the 4A heater closed, the alternate level control valve was unable to maintain level in the 4A heater and extraction steam to the 4A heater closed on HI level. With one low pressure heater string out of service, a down power to 72 percent was required in order to comply with limiting plant conditions specified by OP-003-034, “Feed Heater Vents and Drains.” Following the feedwater heater isolation event, the licensee took action to determine the cause of the unexpected plant conditions. Further troubleshooting revealed that instrumented 5A feedwater heater level was indicating higher than local level sight glass observations in the field. The licensee determined that the function of the 5A feedwater heater level instrument needed to be checked. While implementing a tagout to remove the 5A heater level switch from service, the licensee found the upper isolation valve for the 5A heater level switch closed vice open as expected. The licensee completed functional checks of the 5A feedwater heater level switch with no other abnormalities noted, opened the upper isolation valve for the 5A level switch, and increased the unit back to 100 percent power on December 8, 2009, without incident.

A review of maintenance records indicated that functional checks of the 5A feedwater heater level switch had previously occurred during refueling outage 16 on October 30, 2009 using work order 00180716-01. The functional test required the upper isolation valve for the level switch to be closed during the test. Step 4.4 of work order instructions directed the upper isolation valve to be opened and verified open following completion of the test. A review of tagouts for the associated valve and adjacent equipment showed that no other work was performed between October 30 and December 7. Additionally, a review of the valve deviation log and valve line-ups indicated that no other work or

repositioning of this valve was required or otherwise performed. The licensee performed a root cause evaluation for the event and concluded that the root cause was that the technicians performing the functional test of the 5A feedwater heater on October 30, 2009, failed to perform concurrent verification as required by work order steps to ensure manipulated components were restored to the correct position following maintenance. The immediate corrective actions included restoring the feedwater heater drain valve to its proper position.

Analysis. The performance deficiency was that the licensee's personnel did not implement work instructions as required by the work order. Specifically, the licensee did not perform concurrent verification checks as required by WO-00180716 to ensure that personnel restore and/or manipulate components to the correct position following maintenance. The inspectors determined that this deficiency was reasonably within the ability of the licensee to foresee and prevent occurrence. The finding was more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and affects the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the human error caused an event that upset plant stability during power operation. The inspectors evaluated this finding using Inspection Manual Chapter 0609 Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The inspectors determined that the finding was of very low safety significance (Green) because it does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. In this case, the human error only contributes to the likelihood of a reactor trip. The finding had a cross-cutting aspect in the work practices component of the human performance area because the licensee personnel proceed in the face of uncertainty or unexpected circumstances (H.4(a)).

Enforcement. Enforcement action does not apply because the performance deficiency did not involve a violation of a regulatory requirement. This finding was of very low safety significance and entered into the licensee's corrective action program as CR-WF3-2009-7420. (FIN 05000382/2010005-04: Failure to Implement Work Order Instructions to Restore a Feedwater Heater Drain Valve)

4OA3 Event Follow-up (71153)

.1 (Closed) Licensee Event Report (LER) 05000382/2010001-00, Spent Fuel Pool Cooling Single Failure

On September 13, 2009, during a system review for refueling outage 16, Operation personnel identified a single point vulnerability with a level switch associated with the spent fuel pool (SFP) cooling pumps and SFP purification pumps. The SFP level switch supplies the low level trip and alarm function for both the SFP cooling and purification pumps, respectively. The failure of the level switch or loss of power to the switch would cause all three pumps to trip with no restart capabilities. As a result, this would prevent cooling of the spent fuel pool until the level switch could be repaired or bypassed. The licensee determined that the cause of the condition was a failure to review and evaluate the system's failure modes and effects analysis provided in Table 9.1-4 of the final safety

analysis report, as updated and engineering design calculation ECM98-067, "Limiting Single Failure Thermal-Hydraulic Analysis of Waterford 3 Spent Fuel Pool". The licensee missed opportunities to identify this issue when performing design calculation reviews and component classifications. The corrective actions included the installation of a temporary modification to jumper the level switch such that operators could bypass the switch and restore the fuel pool cooling, if necessary. This licensee-identified finding involved a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control". The finding was more than minor because it was associated with the design control attribute of the Barrier Integrity Cornerstone and affects the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The safety significance and enforcement aspects of the violation are discussed in Section 4OA7. This licensee event report was closed.

.2 (Closed) Licensee Event Report (LER) 05000382/2010003-00, Worn Fuel Oil Line on 'A' Emergency Diesel Generator Caused by Inadequate Mounting Clamp

On February 8, 2010, a nuclear auxiliary operator identified a loose support clamp on the main fuel oil supply line for the 'A' emergency diesel generator (EDG). The operator also noted that the fuel oil supply line tubing under the loose clamp had circumferential wear indications. The licensee initiated a condition report to evaluate circumferential wear indications and determined that the clamp impacted the supply line wall thickness. Specifically, the tube wall thinning was due to wearing of the tubing outer wall by sliding friction between the tube and clamp while the EDG was running. Further investigation discovered that the licensee did not install the mounting clamp per the vendor specifications. As a result, the licensee performed a calculation to determine the tubing wear rate for the 'A' EDG. The licensee concluded that the tube wear rate would have rendered the 'A' EDG inoperable prior to its 30 day mission time. The licensee immediate corrective actions included the replace of the fuel supply line tubing and installed an approve clamp that met vendor specification. This licensee-identified finding involved a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control". The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The safety significance and enforcement aspects of the violation are discussed in Section 4OA7. This licensee event report was closed.

4OA5 Other Activities

.1 (Open) NRC TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01)"

a. Inspection Scope

The inspectors reviewed the licensee's procedures used for filling and venting to verify that the procedures were acceptable for the following: (1) testing the safety injection

system at power operation, shutdown operation, maintenance, and subject system modifications; (2) void determination and elimination methods; and (3) post-event evaluation.

The inspectors reviewed the licensee's procedures used to conduct surveillances and determination of void volumes to ensure that the void criteria satisfied the requirements until the next scheduled void surveillance. The inspectors also reviewed procedure used for filling and venting following conditions which may have introduced voids into the subject systems to verify that the procedures acceptably addressed testing for such voids and provided acceptable processes for their reduction or elimination. Specifically, the inspectors verified that:

- Gas intrusion prevention, refill, venting, monitoring, trending, evaluation, and void correction activities were acceptably controlled by approved operating procedures;

- Procedures ensured the system did not contain voids that may jeopardize operability;

- Procedures established that void criteria were satisfied and will be reasonably ensured to be satisfied until the next scheduled void surveillance;

- The licensee entered changes into the corrective action program as needed to ensure acceptable response to issues. In addition, the inspectors confirmed that a clear schedule for completion of corrective action program that have not been completed;

- Procedures included independent verification that critical steps were completed

The inspectors verified the following with respect to surveillance and void detection:

- Specified surveillance frequency was consistent with technical surveillance requirements;

- Surveillance frequencies were stated or, when conducted more often than required by TSs, the process for their determination was described;

- Surveillances method was acceptably established to achieve the needed accuracy;

- Surveillance procedure included up-to-date acceptance criteria;

- Procedure included effective follow-up actions when acceptance criteria are exceeded or when trending indicates that criteria may be approached before the next scheduled surveillance;

- Measured void volume uncertainty was considered when comparing test data to acceptance criteria;

- Venting procedure and practice utilized criteria such as adequate venting durations and observing a steady stream of water;

- An effective sequencing of void removal steps was followed to ensure that gas does not move into previously filled system volumes;

- Qualitative void assessment methods included expectations that the void will be significantly less that allowed by acceptance criteria;

- Venting results were trended periodically to confirm that the systems are sufficiently full of water and that the venting frequencies are adequate. The inspectors also verified that records on the quantity of gas at each location are maintained and trended as a means of preemptively identifying degrading gas accumulations;

- Surveillances were conducted at any location where a void may form, including high points, dead legs, and locations under closed valves in vertical pipes;

- The licensee ensure that systems were not pre-conditioned by other procedures that may cause a system to be filled, such as by testing, prior to the void surveillance; and

- Procedure included gas sampling for unexpected void increases if the source of the void was unknown and sampling was needed to assist in determining the source.

The inspectors verified the following with respect to filling and venting:

- Revisions to fill and vent procedure to address new vents or different venting sequences were acceptably accomplished; and

- Fill and vent procedure provided instructions to modify restoration guidance to address changes in maintenance work scope or to reflect different boundaries from those assumed in the procedure.

The inspectors verified the following with respect to void control:

- Void removal methods were acceptably addressed by approved procedures; and

- The licensee had reasonably ensured that the high and low pressure safety injection pumps are free of damage following a gas-related event in which pump acceptance criteria was exceeded.

This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later inspection report. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings were identified

.2 (Closed) NRC Temporary Instruction 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event"

a. Inspection Scope

The inspectors assessed the activities and actions taken by the licensee to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. This included (1) an assessment of the licensee's capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis on strategies related to the spent fuel pool, as required by NRC Security Order Section B.5.b issued February 25, 2002, as committed to in severe accident management guidelines, and as required by 10 CFR 50.54(hh); (2) an assessment of the licensee's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63 and station design bases; (3) an assessment of the licensee's capability to mitigate internal and external flooding events, as required by station design bases; and (4) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flood events, which were performed by the licensee to identify any potential loss of function of this equipment during seismic events possible for the site.

b. Findings

Inspection Report 05000382/2011006 (ML11133A162) documented detailed results of this inspection activity. Following issuance of the report, the inspectors conducted detailed follow-up on selected issues. No findings were identified during this follow-up inspection.

.3 (Closed) NRC Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)"

The inspectors reviewed the licensee's severe accident management guidelines (SAMGs), implemented as a voluntary industry initiative in the 1990's, to determine (1) whether the SAMGs were available and updated, (2) whether the licensee had procedures and processes in place to control and update its SAMGs, (3) the nature and extent of the licensee's training of personnel on the use of SAMGs, and (4) licensee personnel's familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant-specific results for Waterford Steam Electric Station, Unit 3 were provided as Enclosure 13 to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated May 27, 2011 (ML111470264).

.4 <u>(Closed) URI 05000382/2010005-01 Foreign Material Exclusion Issue associated with the Condensate Storage Pool Gooseneck Vent</u>

The inspectors opened an unresolved item in NRC Inspection Report 05000382/2010005 to review a potential common mode failure of the emergency feedwater system due to foreign material entering the condensate storage pool from an eight-inch diameter vent line located in the component cooling water pump room of Train 'B'. The inspectors resolved this item after further review of the licensee's administrative controls and evaluation review of the issue once it was entered into the licensee corrective action program. The inspectors determined that adequate controls existed but needed to be enhanced such that it would prevent this potential common mode failure of the emergency feedwater system from occurring. This unresolved item was closed.

4OA6 Meetings

<u>Exit Meeting Summaries</u>

On April 21, 2011, the inspectors presented the inservice inspection results to J. Kowalewski, Vice President, Operations, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any material examined during the inspection should be considered proprietary. No proprietary information was identified.

On April 29, 2011, the inspectors presented the results of the radiation safety inspections to J. Kowalewski, Vice President, Operations, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On May 23, 2011, the inspector conducted a telephonic conference to discuss with Mr. G. Fey, Manager, Emergency Planning, and other members of the licensee staff, the results of in-office inspection of licensee changes to the Emergency Plan and emergency plan implementing procedures. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On July 8, 2011, the inspectors presented the inspection results of the inspection activities to J. Kowalewski, Vice President, Operations, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section 2.3.2 of the NRC Enforcement Policy for being disposition as noncited violations.

.1 Spent Fuel Pool Cooling Single Failure

Criterion III of Appendix B to 10 CFR Part 50 requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, prior to September 13, 2009, the licensee did not verify the adequacy of design basis calculation ECM98-067, "Limiting Single Failure Thermal-Hydraulic Analysis of Spent Fuel Pool," through the performance of a design review. As a result, a single failure of the spent fuel pool level switch would have caused a loss of all the spent fuel pool cooling pumps. The inspectors determined that the finding was of very low safety significance because it did not result in a loss of cooling to the spent fuel pool, whereby operators could preclude restoration of cooling prior to pool cooling, did not result from fuel handling errors that caused damage to fuel clad integrity or dropped assembly, and did not result in a loss of spent fuel pool inventory greater than ten percent of the spent fuel pool volume. The licensee entered this issue into their corrective action program as CR-WF3-2009-4908.

.2 Worn Fuel Oil Line on the 'A' Emergency Diesel Generator Caused by an Inadequate Mounting Clamp

Criterion III of Appendix B to 10 CFR Part 50 requires, in part, that measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to safety-related functions of the structures, systems, and components. Contrary to the above, prior to February 8, the licensee did not select and review the proper clamp for the suitability of the application that was essential to the safety-related function of the EDG fuel oil supply line. As a result, the inadequate clamp would have rendered the EDG inoperable prior to its 30 day mission time. The inspectors determined that the finding was of very low safety significance because it was a design deficiency confirmed not to result in a loss of operability for the probabilistic risk analysis (PRA) mission time of twenty-four hours. The licensee entered this issue into their corrective action program as CR-WF3-2010-0889.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

J. Kowalewski, Vice President, Operations
C. Arnone, General Manager, Plant Operations
C. Alday, Manager, System Engineering
D. Becker, Technical Specialist IV, Programs and Components
E. Begley, Senior Engineer, Programs and Components
D. Boan, Supervisor, Radiation Protection
E. Brauner, Supervisor, System Engineering
J. Brawley, ALARA Supervisor, Radiation Protection
B. Briner, Technical Specialist IV, Programs and Components
A. Buford, Engineer II, System Engineering
K. Cook, Manager, Operations
L. Dauzat, Operations Supervisor, Radiation Protection
C. England, Manager, Radiation Protection
G. Fey, Manager, Emergency Planning
C. Fugate, Assistant Manager, Operations
J. Hashim, Senior Engineer, Programs and Components
M. Haydel, Supervisor, Programs and Components
J. Hornsby, Manager, Chemistry
J. Houghtaling, Senior Project Manager
C. Hunsaker, Code Programs Engineer
H. Landeche, Jr., Senior Technician, Instruments and Controls
B. Lanka, Manager, Design Engineering
B. Lindsey, Manager, Maintenance
M. Mason, Senior Licensing Specialist, Licensing
W. McKinney, Manager, Corrective Action and Assessments
D. Miller, Supervisor, Radiste and Radioactive Material Control
D. Moor, Fleet Manager, Radiation Protection
K. Nichols, Director, Engineering
R. O'Quinn, Steam Generator Program
R. Perry, Senior Emergency Planner
A. Piluti, Manager, Radiation Protection
J. Pollack, Senior Licensing Specialist, Licensing
C. Pramono, Engineer, Systems Engineering
R. Putnam, Manager, Programs and Components
T. Qualantone, Manager, Plant Security
W. Steelman, Manager, Licensing
J. Williams, Senior Licensing Specialist, Licensing

NRC Personnel

M. Davis, Senior Resident Inspector
D. Overland, Resident Inspector
C. Smith, Project Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000382/20110003-05	URI	Loss of Reactor Coolant Inventory during the Assembly of Incore Instrumentation Flanges

Opened and Closed

05000382/2011003-01	NCV	Failure to Evaluate and Adequately Monitor Activities Associated with the Internal Conditions of the Condensate and Refueling Water Storage Pool Structures
05000382/2011003-02	NCV	Failure to Update the FSAR following Modifications to the Reactor Coolant Pump Vapor Seals
05000382/2011003-03	NCV	Failure to Implement Written Procedures for Restoring a Time Delay Relay Associated with the ‘A’ Output Breaker
05000382/2011003-04	FIN	Failure to Implement Work Order Instructions to Restore a Feedwater Heater Drain Valve

Closed

05000382/2010001-00	LER	Spent Fuel Pool Cooling Single Failure
05000382/2010003-00	LER	Worn Fuel Oil Line on the ‘A’ EDG Caused by an Inadequate Mounting Clamp
05000382/20100005-01	URI	Foreign Material Exclusion Issue associated with the Condensate Storage Pool Gooseneck Vent.

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

CONDITION REPORTS

CR-WF3-2011-4093	CR-WF3-2009-1824

WORK ORDERS

261540	275893	275894
231663	231632	186017

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
OP-009-002	Emergency Diesel Generator	313
SD-EDG	Emergency Diesel Generator	4
WF3-SE-08-00001	Summary of Activities Associated with Resolution of GL 2008-01	2
OP-903-001	Technical Specification Surveillance Logs	43
OP-903-030	Safety Injection Pump Operability Verification	18
PE-001-020	Walkdown Process Associated with Managing Gas Accumulation	0
WF3-SE-08-00001	Summary of Activities Associated with Resolution of GL 2008-01	2
EC-14765	SI-405A(B) Bypass Fill/Equalization Line Addition	5/10/2010
STA-001-004	Piping Penetrations	207

DRAWINGS

NUMBER	TITLE	REVISION /DATE
G1114, Sheet 1	Shutdown Cooling Flowpath Through LPSI	12
G167, Sheet 1	Safety Injection System Flow Diagram	49
G167, Sheet 2	Safety Injection System Flow Diagram	52
G167, Sheet 3	Safety Injection System Flow Diagram	20
G167, Sheet 4	Safety Injection System Flow Diagram	17

Section 1R05: Fire Protection

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
UNT-005-013	Fire Protection Program	11
OP-009-004	Fire Protection	307
MM-007-010	Fire Extinguisher Inspection and Replacement	304
FP-001-015	Fire Protection System Impairments	303
OP-903-060	Fire Hose Station Inspection	8

Section 1R06: Flood Protection

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
MNQ3-5	Flooding Analysis Outside Containment	4
EC-M99-010	Dry Cooling Tower Basin Ponding Analysis	0

Section 1RO8: Inservice Inspection Activities

CONDITION REPORTS

CR-WF3-2009-5742	CR-WF3-2009-6032	CR-WF3-2009-6486	CR-WF3-2009-6054
CR-WF3-2009-6514	CR-WF3-2009-6622	CR-WF3-2009-6629	CR-WF3-2009-6883
CR-WF3-2010-0352	CR-WF3-2010-4328	CR-WF3-2010-4897	CR-WF3-2010-5071
CR-WF3-2011-2423	CR-WF3-201102450	CR-WF3-2011-2463	CR-WF3-2011-2578
CR-WF3-2011-2616	CR-WF3-2011-2642	CR-WF3-2009-5112	CR-WF3-2009-6995
CR-WF3-2009-6995	CR-WF3-2010-5530	CR-WF3-2010-5696	CR-WF3-2010-1390
CR-WF3-2010-1300	CR-WF3-2010-5739	CR-WF3-2010-3378	CR-WF3-2010-3372
CR-WF3-2010-3235	CR-WF3-2010-0041	CR-WF3-2011-1970	CR-WF3-2011-2389
CR-WF3-2011-1971	CR-WF3-2011-1972	CR-WF3-2011-1491	CR-WF3-2011-1969
CR-WF3-2011-2022	CR-WF3-2011-2024	CR-WF3-2011-2053	CR-WF3-2011-1492
CR-WF3-2011-1687	CR-WF3-2011-2048	CR-WF3-2011-1491	CR-WF3-2011-2105
CR-WF3-2011-2148	CR-WF3-2011-2181	CR-WF3-2008-4890	

WORK ORDERS

116193	221792	222443	222444
222448	222449	222511	259868
222446			

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION /DATE
WDI-STD-1041	Reactor Vessel Head Penetration Ultrasonic Examination Analysis	4
URS-UT-PDI-8	Manual Ultrasonic Examination of Weld Overlaid Similar & Dissimilar Metal Welds	0
EN-LI-102	Corrective Action Process	November 1, 2010
CEP-NDE-0497	Manual Ultrasonic Examination of Welds in Vessels	5
E-P1-A-A1	Welding Procedure Specification (WPS)	1
E-P8-T-A8, Ar	Welding Procedure Specification (WPS)	0
CEP-NDE-0485	Manual Ultrasonic Examination of Vessel Nozzle Inside Radius	6
PDI-UT-8	Generic Procedure for the Ultrasonic Examination of Weld Overlay Similar and Dissimilar Metal Welds	F
NOECP-107	Boric Acid Corrosion Control Program	3
EN-DC-319	Inspection and Evaluation of Boric Acid Leaks	6
CEP-NDE-0955	Visual Examination (VE) of Bare-Metal Surfaces	302
PDI-UT-8	Generic Procedure for Ultrasonic Examination of Weld Overlaid Similar and Dissimilar Metal Welds	F
EN-LI-119	Apparent Cause Evaluation (ACE) Process	8
EN-DC-317	Steam Generator Program	5
HC-00.ET-I	Specific Procedure for Training, Qualification and Certification of Personnel Participating in Steam Generator Tubes Eddy Current Inspections (TECNATOM)	9
HC-00	Qualification and Certification of Non-Destructive Testing Personnel (TECNATOM)	15
SAEC-QAP 9.1	NDE Personnel Qualification and Certification	18
CC-A-22	Qualification and Certification of Personnel in Nondestructive Testing (KPS)	16
ANATEC-08	Certification of NDT Personnel (Eddy Current Method)	22
LTR-SGDA-11-47	Requirements for Inspection of the Internal Feedwater Piping Assembly at Waterford 3 during RF17	April 4, 2011

LTR-CDME-08-30	Assessment of Operational Leakage Potential for Sentinel Plugged Tubes: Waterford 3	January 6, 2011
LTR-SGDA-11-70	Updated Acceptance Criteria of Intrados Tube Wear at Waterford Unit 3	March 31,2011
CEP-ISI-001	Waterford 3 Steam Electric Station Inservice Inspection Plan	307
WEC 2.10	Qualification, Training and Certification of Nondestructive Personnel	1
SG-SGMP-11-5	SG Degradation Assessment for Waterford 3 Nuclear Plant RF17 Refueling Outage	March 31, 2011
NOECP-252	Steam Generator Eddy Current Inservice Testing	12
CWTR3-SG-001	Standard In Situ Pressure Test Using the Computerized Data Acquisition System	5
W3F1-2010-0014	Response to NRC Request for Additional Information dated January 6, 2010 Re: Waterford 3 Steam Electric Station, Unit 3 – Requests for Relief for ASME Section XI Volumetric Examination Requirements – Second 10 Year Inservice Inspection Interval – Waterford Steam Electric Station, Unit 3	February 8, 2010
NRC Letter	Waterford Steam Electric Station, Unit 3 – Request for Relief Nos. WF3-ISI-007, WF3-ISI-008, WF3-ISI-010, WF3-ISI-011, WF3-ISI-012, WF3-ISI-013, and WF3-ISI-014 from ASME Code, Section XI, Examination Requirements for Second 10 – Year Inservice Inspection Interval	June 30, 2010
STD-400-173	Checkout and Operation of the Steam Generator Tube Standard In Situ Pressure Testing System	13
URS-UT-PDI-8	UT Calibration/Examination	0

DRAWINGS

NUMBER	TITLE	REVISION
E-9270-164-005	Nozzle Requirements Closure Head	7

MISCELLANEOUS DOCUMENT

NUMBER	TITLE	REVISION /DATE
WCAP-15988	EPRI Guidance for Effective Monitoring Boric Acid Inspection Program	1
W3F1-2010-0033	Response to the NRC Request for Additional Information Re: Waterford Steam Electric Station, Unit 3 – Requests for Relief WF3-ISI-007, 008, 009, 010, 011, 012, 013, and 014	April 29, 2010
W3F1-2011-0014	Request for Alternative W3-ISI-019, Inspection of Reactor Vessel Head In-Core Instrument Nozzles during the Third Ten –Year Inservice Inspection Interval	February 16, 2011
W3F1-2011-0013	Request for Alternative W3-ISI-018, Inspection of Reactor Pressure Vessel Head Control Element Drive Mechanism Nozzles during the Third Ten-Year Inservice Inspection Interval	February 16, 2011
STD-400-173	Checkout and Operation of the Steam Generator Tube Standard In Situ Pressure Test System	13
PQR 107	Procedure Qualification Record	1
PQR 024	Procedure Qualification Record	1
PQR 170	Procedure Qualification Record	1
PQR 029	Procedure Qualification Record	1
PQR 330	Procedure Qualification Record	1
PQR 331	Procedure Qualification Record	1
ASME Code Case N-729-1	Alternative Examination Requirements for PWR Reactor Vessel Upper Head N-722	March 28, 2006

NONDESTRUCTIVE EVAULATION REPORTS

ISI-UT-11-003	ISI-UT-11-002	ISI-VE-11-001	ISI-VE-11-002
ISI-VE-11-003	ISI-VE-11-004	ISI-VE-11-005	ISI-VE-11-006
ISI-VE-11-007	ISI-VE-11-008	ISI-VE-11-009	

Section 1R11: Licensed Operator Requalification Program

PROCEDURES/DOCUMENTS

NUMBER	TITLE	Revision
P-138	Simulator Scenario for Waterford 3 Nuclear Plant	0

Section 1R12: Maintenance Effectiveness

CONDITION REPORTS

CR-WF3-2011-1168 CR-WF3-2004-0781

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION /DATE
EN-DC-203	Maintenance Rule Program	1
EN-DC-150	Condition Monitoring of Maintenance Rule Structures	1
EN-DC-153	Preventive Maintenance Component Classification	5
LO-NOE-2004-0067	Operating Experience Impact Evaluation for NRC IN-04-005	March 16, 2004
NRC Information Notice 2004-05	Spent Fuel Pool Leakage to Onsite Groundwater	March 3, 2004
NRC Generic Issue No. 202	Spent fuel Pool Leakage Limits	August 2006
W-CS-2003-001-00	Maintenance Rule Walkdown for Evaluation of Structures	0
WF3-CS-11-00001	Maintenance Rule Walkdown for Evaluation of Structures	0
NUREG – 1522	Assessment of Inservice Condition of Safety-Related Nuclear Power Plant Structures	June 1995
NUREG/CR-6715	Probability-Based Evaluation of Degraded Reinforced Concrete Components in Nuclear Power Plants	March 2001
ACI Standard 349.3R	Evaluation of Existing Nuclear Safety-Related Concrete Structures, American Concrete Institute	1992
WF3-SE-08-00002	Waterford 3 Maintenance Rule 10CFR50 (a)(3) Periodic Assessment Cycle 15	0

DRAWINGS

NUMBER	TITLE	REVISION
G134	General Arrangement Reactor Auxiliary Bldg. Plan EL +46.00’	35

G135	General Arrangement Reactor Auxiliary Bldg. Plan EL +21.00’	30
G136	General Arrangement Reactor Auxiliary Bldg. Plan EL -4.00’	35
G137	General Arrangement Reactor Auxiliary Bldg. Plan EL -35.00’	27
G138	General Arrangement Reactor Auxiliary Building – Section Sheet 1	20
G907	Reactor Auxiliary Bldg Pool Liner Details	9
G906	Reactor Auxiliary Bldg Condensate Pool Liner	3
G905	Reactor Auxiliary Bldg Refuel Pool Liner	4

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

CONDITION REPORTS

CR-WF3-2011-4093	CR-WF3-2011-4448	CR-WF3-2011-3599	CR-WF3-2011-3600

WORK ORDERS

261540	275893	275894	278784
281799	282018		

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
EN-WM-101	On-line Work Management Process	6
OI-037-000	Operations’ Risk Assessment Guideline	2
OP-009-002	Emergency Diesel Generator	313
SD-EDG	Emergency Diesel Generator	4
OP-002-004	Chilled Water System	305
SD-CHW	Essential Chilled Water	6

Section 1R15: Operability Evaluations

CONDITION REPORTS

CR-WF3-2011-4227	CR-WF3-2011-3600	CR-WF3-2011-1877

WORK ORDERS

WR 232426

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
EN-OP-104	Operability Determination Process	4
EN-WM-101	On-Line Work Management Process	6
OI-037-000	Operations Risk Management Guideline	300
OP-100-010	Equipment Out of Service	303
W2.502	Configuration Risk Management Program Implementation	0
OP-009-002	Emergency Diesel Generator	313
SD-EDG	Emergency Diesel Generator	4

Section 1R18: Plant Modifications

CONDITION REPORTS

CR-WF3-2007-3716	CR-WF3-2009-5501	CR-WF3-2009-5822	CR-WF3-2009-5884
CR-WF3-2007-3716	CR-WF3-2007-3716	CR-WF3-2010-7466	CR-WF3-2010-7421
CR-WF3-2011-0553	CR-WF3-2011-1965		

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
EC-06256	Reactor Coolant Pump Vapor Stage Leakage Reroute to Floor	0
EC-18520	Reactor Coolant Pump Vapor Stage Leak-off Line & Trough Modification	0
FSAR Chapter 5	Waterford 3 Final Safety Analysis Report, Reactor Coolant System and Connected Systems	304

Section 1R19: Post Maintenance Testing

CONDITION REPORTS

CR-WF3-2011-4093	CR-WF3-2011-2104	CR-WF3-2011-2665	CR-WF3-2011-2733
CR-WF3-2011-2901	CR-WF3-2011-2978	CR-WF3-2011-3190	CR-WF3-2011-3219

WORK ORDERS

261540	275893	275894	210424
219354	WR#234065	273846	274481
275294-01	52230980-01	75061-01	

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
OP-009-002	Emergency Diesel Generator	313
SD-EDG	Emergency Diesel Generator	4
OP-903-068	Emergency Diesel Generator Operability and Subgroup Relay Operability Verification	303
ME-003-200	Station Battery Bank & Charger (Weekly)	306
OP-903-115	Train A Integrated Emergency Diesel Generator/Engineering Safety Features	15
ME-007-005	Time Delay Relay Setting Check, Adjustment, and Functional Test	15
MG-33	Configuration Control Guidelines and Completing Lifted Lead & Switch Manipulation Forms	12
EN-WM-105	Planning	9
EN-WM-107	Post Maintenance Testing	3

DRAWINGS

NUMBER	TITLE	REVISION
B424, sheet 2327	Diesel Generator 'A' Breaker	16
5817-3193	Auxiliary Panel 1DD Wiring Diagram	30

Section 1R20: Refueling Outage

CONDITION REPORTS

CR-WF3-2011-2987	CR-WF3-2011-2971	CR-WF3-2011-2201	CR-WF3-2011-2209
CR-WF3-2011-3163	CR-WF3-2011-3350	CR-WF3-2011-3636	

WORK ORDERS

247681	247682	248838-04

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
OP-010-003	Plant Start-up	320
OP-010-004	Power Operations	313
OP-010-005	Plant Shutdown	314
OP-010-006	Outage Operations	314

OP-001-001	Reactor Coolant System Fill and Vent	25
OP-001-003	Reactor Coolant System Drain Down	308
OP-001-005	RCS Drain and Fill Below RCS Hot Leg Centerline	306
PLG-009-014,	Conduct of Planned Outages	305
OI-041-000	Operations Outage Guide	6
RF-005-001	Fuel Movement	307
RF-001-013	Incore Instrument Flanges	304
OP-903-067	Unit Power Supply Transfer Check	8

Section 1R22: Surveillance Testing

CONDITION REPORTS

CR-WF3-2011-4093	CR-WF3-2011-1419	CR-WF3-2011-2933

WORK ORDERS

261540	275893	275894	277145

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
OP-903-068	Emergency Diesel Generator Operability and Subgroup Relay Operability Verification	303
SD-EDG	Emergency Diesel Generator	4
STA-001-004	Local Leak Rate Test (LLRT)	308
OP-903-033	Cold Shutdown IST Valve Test	34

Section 2RS01: Radiological Hazard Assessment and Exposure Controls

CONDITION REPORTS

CR-WF3-2011-00037	CR-WF3-2011-00038	CR-WF3-2011-00039	CR-WF3-2011-03037
CR-WF3-2011-03017	CR-WF3-2011-03036	CR-WF3-2011-03037	CR-WF3-2011-03083
CR-WF3-2011-03095	CR-WF3-2011-03141		

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
EN-RP-100	Radiation Worker Expectations	6

EN-RP-101	Access Control for Radiologically Controlled Areas	5
EN-RP-102	Radiological Control	2
EN-RP-105	Radiological Work Permits	9
EN-RP-108	Radiation Protection Posting	9
EN-RP-121	Radioactive Material Control	6
EN-RP-131	Radioactive Material Control	8
EN-RP-143	Source Control	7
EN-RP-202	Personnel Monitoring	7
EN-RP-203	Dose Assessment	4
EN-RP-204	Special Monitoring Requirements	3
EN-RP-205	Prenatal Monitoring	3
EN-RP-402	DOP Challenge Testing of HEPA Vacuums and Portable Ventilation Systems	4
EN-RP-404	Operation and Maintenance of HEPA Vacuums and Portable Ventilation Units	4
HP-002-222	Steam Generator Radiological Controls	8

RADIATION WORK PERMITS

20110508	Inspect/Rework RCP Motors 1A, 1B, 2A, 2B
20110509	Remove/Replace Steam Generator Primary Manways
20110510	Install/Remove Steam Generator Nozzle Dams
20110511	Perform Eddy Current Work/Tube Plugging Inside of the Steam Generators Primary Side
20110600	Radiation Protection

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

Number	TITLE	DATE
LO-WLO-2010-00025-CA-00001	Occupational Radiation Safety/ALARA Pre-NRC Inspection	February 25, 2010
QS-2011-W3-327	QA Follow-up Surveillance of 02C-W3-2010-0327 Cobalt Reduction	February 21, 2011

MISCELLANEOUS DOCUMENTS

	TITLE	DATE
	Radioactive Source List	December 20, 2010
	Refuel Outage 16 Radiation Protection Report	

NSTS Annual Inventory Reconciliation	January 5, 2011
Spent Fuel Storage Rack Inventory Map	April 7, 2011

Section 2RS02: Occupational ALARA Planning and Controls

CONDITION REPORTS

CR-WF3-2010-07131	CR-WF3-2010-07530	CR-WF3-2010-07625	CR-WF3-2011-00037
CR-WF3-2011-00418	CR-WF3-2011-00448		

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
EN-RP-110	ALARA Program	7
EN-RP-110-01	ALARA Initiative Deferrals	0
EN-RP-110-02	Elemental Cobalt Sampling	0
EN-RP-141	Job Coverage	5
EN-FAP-RP-001	Corporate ALARA Committee	1
HP-001-114	Control of Temporary Shielding	11
HP-001-150	Use of Protective Clothing	13

RADIATION WORK PERMITS

NUMBER	TITLE
20110511	To Perform Eddy Current Work/Tube Plugging Inside of the Steam Generators Primary Side
20110515	Remove/Install RCP 1B and 2A Seal, Rotating Baffle, Detension Heat Exchanger fasteners and Perform Visual Inspection
20110620	Install/Remove Temporary Shielding in Radiation and High Radiation Areas of the Reactor Containment Building
20110702	Disassembly of Reactor Head and All Associated Work Activities

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
LO-WLO-2010-00025-CA-00001	RP Occupational Radiation Safety/ALARA Pre-NRC Inspection	February 25, 2010

QS-2011-W3-02	QA Follow-up Surveillance of 02C-W3-2010-0327 Cobalt Reduction Program	February 21, 2011

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
	2009-2013 WF3 Five Year ALARA Plan	April 27, 2009
	Waterford-3 Dose Saving Initiatives Chart	
	Refuel 17 Primary Chemistry Plan	January 31, 2011
	Waterford-3 Operations Shift Outage Turnover Report	April 26-29, 2011
	Refuel 17 Reactor Coolant System Co-58 Cleanup	April 10, 2011
	ALARA Manager's Committee Meeting Minutes	Various Dates in April 2011
EC-0021253/ 0021287	Engineering Change / 50.59 Evaluation for RCP Seal Drain Collection System Installation	April 22, 2010

Section 2RS03: In-Plant Airborne Radioactivity Control and Mitigation

CORRECTIVE ACTION DOCUMENTS

CR-WF3-2010-03306 CR-WF3-2011-02000

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
EN-RP-501	Respiratory Protection Program	4
EN-RP-502	Inspection and Maintenance of Respiratory Protection Equipment	6
EN-RP-503	Selection, Issue and Use of Respiratory Protection Equipment	5
EN-RP-504	Breathing Air	3
EN-RP-505	Portacount Respirator Fit Testing	2
HP-002-630	Verification of Breathing Air Quality	9

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

Number	TITLE	DATE
LO-WLO-2010-00040	Radiation Protection – Respiratory Protection Program Effectiveness	April 29, 2010

MISCELLANEOUS DOCUMENTS

TITLE	DATE
SCBA Functional Tests	December 2009 through March 2011
SCBA Cylinder Hydrostatic Tests	November 15, 2010 through March 7, 2011
SCBA Cylinder Grade D / Grade L Analysis Results	Various Dates in 2010 and 2011
Portable Instrumentation Calibration Data Sheets: Respirators	Various Dates in 2010 and 2011
MSA Certification for Entergy Operations, Inc. of Port Gibson, MS (Grand Gulf Nuclear Station)	September 15, 2010
Respiratory Protection Training records	Various Dates from 2008 thru 2011

Section 4OA1: Performance Indicator

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
NEI 99-02	Regulatory Assessment Performance Indicator Guideline	6
EN-LI-114	Performance Indicator Process	4
EN-EP-201	Performance Indicators	9, 10
EP-001-001	Recognition and Classification of Emergency Conditions	24, 25
EP-002-010	Notifications and Communications	303, 304
EP-002-052	Protective Action Guidelines	20, 21
	Waterford3 Steam Electric Station Emergency Plan	38, 39
EN-FAP-RP-002	Radiation Protection Performance Indicator Program	0

Section 4OA2: Problem Identification and Resolution

CONDITION REPORTS

CR-WF3-2009-7420

WORK ORDERS

180716	218505	119546

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
EN-LI-102	Corrective Action Process	16

EN-LI-118	Root Cause Analysis Process	13
OP-003-034	Feed Heater Vents and Drains	6
OP-500-001	Annunciator Response Cabinet A	19
UNT-005-010	Verification Guidance	10

DRAWINGS

NUMBER	TITLE	REVISION /DATE
G155, Sheet 1	Flow Diagram of the Heater Drain & Vent System	36
G155, Sheet 2	Flow Diagram of the Heater Drain & Vent System	24
G155, Sheet 3	Flow Diagram of the Heater Drain & Vent System	30
G155, Sheet 4	Flow Diagram of the Heater Drain & Vent System	5

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion

CONDITION REPORTS

CR-WF3-2010-0889 CR-WF3-2010-0938 CR-WF3-2009-04908

WORK ORDERS

240277

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
ECM98-067	Limiting Single Failure Thermal-Hydraulic Analysis of Waterford 3 Spent Fuel Pool	1
OP-903-068	EDG Operability and Subgroup Operability Verification	303
OP-901-513	Spent Fuel Pool Cooling Malfunction	5
EC 18232	Fuel Pool Pumps A&B Water Level Switch Bypass	1
W3F1-2010-0043	LER 2010-003-00, Worn Fuel Oil Line on A Train Emergency Diesel Generator Caused by Inadequate Mounting Clamp	0
W3F1-2010-0013	LER 2010-001-00, Spent Fuel Pool Single Failure	0

Section 4OA5: Other Activities

CONDITION REPORTS

CR-WF3-2009-1824

WORK ORDERS

231663	231632	186017

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
WF3-SE-08-00001	Summary of Activities Associated with Resolution of GL 2008-01	2
OP-903-001	Technical Specification Surveillance Logs	43
OP-903-030	Safety Injection Pump Operability Verification	18
PE-001-020	Walkdown Process Associated with Managing Gas Accumulation	0
EC-14765	SI-405A(B) Bypass Fill/Equalization Line Addition	May 10, 2010
OP-009-008	Safety Injection System	30