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

April 5, 2010

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 05000382/2009005 ERRATA

Dear Mr. Kowalewski:

Please replace the subject integrated inspection report, ML Number 100360782, dated February 5, 2010, with the attached errata inspection report. The attached errata report contains a revision that was necessary to properly document a biennial licensed operator requalification inspection that was completed on November 20, 2009.

On December 31, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Waterford Steam Electric Station, Unit 3. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 11, 2010, 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.

This report documents three self-revealing findings of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is 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 VI.A.1 of the NRC Enforcement Policy. If you contest the violations or the significance 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 E 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

Inspector at the Waterford Steam Electric Station, Unit 3 facility. In addition, if you disagree with the characterization of 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 Resident Inspector at Waterford Steam Electric Station, Unit 3. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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

Sincerely,

/RA/

Jeffrey A. Clark, P.E.
Chief, Project Branch E
Division of Reactor Projects

Docket: 50-382
License: NPF-38

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NRC Inspection Report 05000382/2009005
w/Attachment: Supplemental Information

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ADAMS: ☐ No ☒ Yes	☒ SUNSI Review Complete	Reviewer Initials: JAC
	☒ Publicly Available	☒ Non-Sensitive
	☐ Non-publicly Available	☐ Sensitive

DRS/OB	C:DRS/OB	C:DRP/E		
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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	05000382
License:	NFP-38
Report:	05000382/2009005
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, LA
Dates:	October 8 through December 31, 2009
Inspectors:	M. Haire, Senior Resident Inspector D. Overland, Resident Inspector S. Anderson, General Engineer R. Azua, Senior Project Engineer M. Bloodgood, Senior Reactor Inspector T. Buchanan, Reactor Inspector S. Garchow, Senior Operations Engineer C. Steely, Operations Engineer T. Pate, Operations Engineer P. Elkmann, Senior Emergency Preparedness Inspector L. Ricketson, P.E., Senior Health Physicist N. Greene, Health Physicist
Approved By:	Jeff Clark, P.E., Chief, Project Branch E Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000382/2009005; October 8, 2009 through December 31, 2009; Waterford Steam Electric Station, Unit 3, Identification and Resolution of Problems, Access Control to Radiologically Significant Areas

The report covered a 3-month period of inspection by resident inspectors and announced baseline inspections by regional based inspectors. Three Green noncited violations of NRC requirements were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." 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

- Green. A self-revealing Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, was identified for the licensee's failure to promptly correct a condition adverse to quality. Specifically, the licensee did not promptly correct reactor coolant pump vapor seal leakage that resulted in boric acid accumulation on the component cooling water heat exchanger and cover areas of three reactor coolant pumps. Corrective actions for this condition were implemented during Refueling Outage 15, but these corrective actions failed to correct the condition and the vapor seal leakage continued through operating Cycle 16. This resulted in some additional boric acid corrosion and degradation to reactor coolant pump covers and carbon steel component cooling water flanges. The licensee implemented a design modification to correct the condition and documented the condition in Condition Report CR-WF3-2009-5501.

  The licensee's failure to promptly correct a condition adverse to quality is more than minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and affects the cornerstone objective to limit the likelihood of those events that upset plant stability. The finding has very low safety significance because, although the finding contributes to the likelihood of a reactor trip, mitigation equipment was still available. This finding had a crosscutting aspect in the area of human performance associated with work control in that the licensee did not effectively plan for the resources necessary to implement the postmaintenance testing associated with the corrective actions implemented during Refueling Outage 15, and therefore failed to discover that those corrective actions were inadequate to correct the condition [H.3(a)] (Section 4OA2).

- Green. A self-revealing Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, was identified for the licensee's failure to prescribe an activity affecting quality by documented instructions, procedures, or drawings appropriate to the circumstance. Specifically, for all reactor coolant pump heat exchanger to pump cover bolted connection gasket replacements between the refueling outage of 1986 (Refueling Outage 1) and the refueling outage of 2009 (Refueling Outage 16), the licensee prescribed the wrong gasket material, gasket size, and fastener preload because they had failed to incorporate a design change implemented during Refueling Outage 1 into their instructions, procedures, or drawings. Station Modification Package SMP-1427, an engineering change implemented during Refueling Outage 1 in response to industry operating experience, called for a thicker gasket, different gasket material, and an increased bolt preload in order to increase gasket compression and reduce the probability of leakage. As a consequence of failing to incorporate Station Modification Package SMP-1427 changes into procedures, all heat exchanger gasket replacements since Refueling Outage 1, four gasket replacements in total, have utilized thinner gaskets with less than the vendor recommended compression. The licensee documented this condition in Condition Report CR-WF3-2009-5501.

  The licensee's failure to prescribe appropriate gasket replacement requirements is more than minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and affects the cornerstone objective to limit the likelihood of those events that upset plant stability. The finding has very low safety significance because, although the finding contributes to the likelihood of a reactor trip, mitigation equipment is still available. This finding had a crosscutting aspect in the area of problem identification and resolution associated with operating experience in that the licensee did not institutionalize operating experience through changes to the station procedures [P.2(b)] (Section 4OA2).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors reviewed a self-revealing noncited violation of Technical Specification 6.8.1 which resulted from a worker failing to follow radiation protection procedures. A contract radiation worker went to work near steam generator 1 rather than the area for which he/she was briefed and received multiple electronic dosimeter dose rate alarms, but did not leave the area until receiving a continuous dose alarm. In response, the licensee investigated the occurrence and restricted the individual's access. Additional actions were being evaluated. This issue was entered into the licensee's corrective action program as Condition Reports CR-WF3-2009-05648 and WF3-2009-06852.

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  This finding is greater than minor because it involved the program attribute of exposure control and affected the cornerstone objective in that the failure of the worker to follow procedural guidance resulted in the worker being unknowledgeable to the dose rates in all areas entered. The inspectors used the

Occupational Radiation Safety Significance Determination Process and determined the finding had very low safety significance because it was not: (1) an as low as reasonably achievable (ALARA) finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an inability to assess dose. The finding had a crosscutting aspect in the area of human performance, work practices component, because the worker failed to use human error prevention techniques such as self and peer checking [H.4(a)] (Section 2OS1).

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking numbers (condition report numbers) are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period on October 8, 2009, at 100 percent power and remained at approximately 100 percent power until October 19, 2009, when the plant was shutdown in preparation of the licensee's planned Refueling Outage 16. The plant remained shutdown until December 1, 2009, when the reactor was placed back online and the licensee began increasing power. On December 6, 2009, the plant reached 100 percent power and continued to operate at this level for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 Readiness to Cope with External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Final Safety Analysis Report for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site that would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also reviewed the abnormal operating procedure for mitigating the design basis flood to ensure it could be implemented as written. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one external flooding sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

- October 8, 2009, Essential chiller train B
- October 14, 2009, Low pressure safety injection train B

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 walked down 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 two partial system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings of significance 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:

- November 2, 2009, Fuel handling building
- November 10, 2009, Fire zones RAB 37, 38, and 39
- November 28, 2009, Reactor containment building
- December 15, 2009, Battery and switchgear areas

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 quarterly fire-protection inspection samples as defined in Inspection Procedure 71111.05-05.

c. Findings

No findings of significance 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 walked down the area 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.

- October 14, 2009, Reactor Auxiliary Building -35 foot elevation

This inspection procedure also requires an annual review of risk-significant cables located in underground bunkers/manholes. Waterford Steam Electric Station, Unit 3, by design, does not have any safety-related cables that are located in underground bunkers/manholes; however, there are 17 manholes in which cables associated with maintenance rule related equipment were located. The inspectors inspected Manholes M301-NA, M346-NB, and M347-NA and determined that all three contained maintenance rule related cables submerged in water. The submerged cables did not show visible deterioration. The licensee has documented this condition in Condition Report CR-WF3-2009-3925, and is developing a cable monitoring program. Specific documents reviewed during this inspection are listed in the attachment.

This activity constitutes completion of two flood protection measures inspection samples as defined in Inspection Procedure 71111.06-05.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the steam generators. The inspectors verified that performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; the licensee utilized the periodic maintenance method outlined in EPRI Report NP 7552, “Heat Exchanger Performance Monitoring Guidelines”; the licensee properly utilized biofouling controls; the licensee’s heat exchanger inspections adequately assessed the state of cleanliness of their tubes; and the heat exchanger was correctly categorized under 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one heat sink inspection sample as defined in Inspection Procedure 71111.07-05.

b. Findings

No findings of significance were identified.

1R08 In-service Inspection Activities (71111.08)

Completion of Sections .1 through .5, below, constitutes completion of one sample as defined in Inspection Procedure 71111.05-05.

.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 reviewed two types of nondestructive examination activities and two welds on the reactor coolant system pressure boundary.

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Safety Injection System	RCS 2A Safety Injection Nozzle (Weld No. 12-009)	Ultrasonic Testing
Reactor Coolant System	RCS 1A Cold leg Suction Line (Weld No. 07-005)	Ultrasonic Testing
Reactor Coolant System	RCS 2A Cold Leg Suction Line (Weld No. 11-002)	Ultrasonic Testing
Reactor Coolant System	RCS 2A Cold Leg Suction Line (Weld No. 11-002)	Visual Inspection VT-1&2

The inspectors reviewed records for the following nondestructive examinations:

SYSTEM	IDENTIFICATION	EXAMINATION TYPE
Safety Injection System	RCS 2A Safety Injection Nozzle (Weld No. 12-009)	Ultrasonic Testing
Reactor Coolant System	RCS 1A Cold leg Suction Line (Weld No. 07-005)	Ultrasonic Testing
Reactor Coolant System	RCS 2A Cold Leg Suction Line (Weld No. 11-002)	Ultrasonic Testing
Reactor Coolant System	RCS 2A Cold Leg Suction Line (Weld No. 11-002)	Visual Inspection VT-1&2
Reactor Coolant System	RCS 12” Hot Leg Surge Line (Weld No.15-009)	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 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 basis 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 of significance 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.

The inspectors also reviewed the results of licensee personnel's volumetric inspection of pressure-retaining components above the reactor pressure vessel head to verify that there were no flaws in the welds associated with these penetrations. The inspectors observed data acquisition and analysis of one penetration. The inspector verified that the personnel performing the inspections were current in their certification as Level II or Level III ultrasonic testing 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 of significance 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 1. 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 of significance 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. No conditions were identified that warranted in-situ pressure testing. The inspectors did, however, review the licensee's "Steam Generator Degradation Assessment and Repair Criteria for RF15," dated April 2008, and compared the in-situ test screening parameters to the guidelines contained in the EPRI document "In Situ Pressure Test Guidelines," Revision 2. This review determined that the screening parameters were consistent with the EPRI guidelines.

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 of significance were identified.

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

a. Inspection scope

The inspectors reviewed 27 condition reports which dealt with inservice inspection activities and found the corrective actions 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 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 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 of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Biennial Inspection

a. Inspection Scope

To assess the performance effectiveness of the licensed operator requalification program, the inspectors conducted personnel interviews, reviewed both the operating tests and written examinations, reviewed randomly selected medical and watchstanding proficiency records, and observed ongoing operating test activities.
The on-site inspection effort occurred from July 13 through 17, 2009. During this time, the inspectors interviewed licensee personnel to determine their understanding of the policies and practices for administering requalification examinations. The inspectors also reviewed operator performance on the periodic written exams and annual operating tests. These reviews included observations of portions of the operating tests by the

inspectors. The operating tests observed included six job performance measures and two scenarios that were used in the current biennial requalification cycle. These observations allowed the inspectors to assess the licensee's effectiveness in conducting the operating test to ensure operator mastery of the training program content.

The results of these examinations were reviewed to determine the effectiveness of the licensee's appraisal of operator performance and to determine if feedback of performance analyses into the requalification training program was being accomplished. The inspectors interviewed members of the training department and reviewed minutes of the Training Oversight Committee to assess the responsiveness of the licensed operator requalification program to incorporate the lessons learned from both plant and industry events. The inspector also reviewed a sample of licensed operator annual medical forms and procedures governing the medical examination process for conformance to 10 CFR 55.53, a sampling of the licensed requalification program feedback system, and the remediation process records.

In addition to the above, the inspectors reviewed examination security measures, simulator fidelity, and existing logs of simulator deficiencies.

The inspectors performed an in-office review of the overall pass/fail results of the individual job performance measure operating tests, simulator operating tests, and written examinations administered by the licensee during the operator licensing requalification cycles and biennial examination. Final examination results were assessed to determine if they were consistent with the guidance contained in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors", Revision 9, Supplement 1, and NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process." Nine separate crews participated in simulator operating tests, written examinations, and job performance measure operating tests, totaling 36 licensed operators. There were no failures on the written examination, simulator operating tests, or job performance measure operating tests.

The inspectors completed one inspection sample of the biennial licensed operator requalification program.

b. Findings

No findings of significance were identified.

.2 Quarterly Inspection

a. Inspection Scope

On November 24, 2009, 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 licensee 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
- Control board manipulations
- Oversight and direction from supervisors

The inspectors compared the crew's performance in these areas to pre-established operator action expectations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one quarterly licensed-operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

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

- November 20, 2009, Effects of voiding on the functionality of low pressure safety injection system

- December 8, 2009, Effects of excessive leakage on functionality of containment isolation valves

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 quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings of significance were identified.

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:

- October 24, 2009, Scheduled plant refuel outage with reactor coolant system water level reduced to approximately 19 feet to support reactor vessel head removal during mode 6 operations

- November 30, 2009, Scheduled activity to take the reactor coolant system solid and draw a bubble in the pressurizer following the refueling outage

- December 13, 2009, Scheduled plant protection system channel B functional test

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 three maintenance risk assessments and emergent work control inspection samples as defined in Inspection Procedure 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- October 29, 2009, Log power nuclear instrument channel B
- November 16, 2009, Station battery train B total allowable resistance
- November 19, 2009, Broken in-core nuclear instrumentation E-13

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 Safety Analysis Report to the licensee’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 operability evaluations inspection samples as defined in Inspection Procedure 71111.15-05.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following postmaintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- November 9, 2009, S6X (41 second load block relay for emergency diesel generator B sequencer) loose terminal adjustments retested during Operating Procedure OP-903-116
- November 10, 2009, Removal, inspection, stroke test, and re-installment 3 plus a safety injection sump outlet header B check valve SI-604B
- November 17, 2009, Replacement of station battery 3-AB-S due to end of useful life
- November 19, 2009, Adjustment to closing force for reactor coolant loop 1 shutdown cooling outside containment isolation valve SI-407B to correct excessive leakage
- November 30, 2009, Emergency feedwater pump AB operability check (Operating Procedure OP-903-046)
- December 7, 2009, Replacement of station battery 3-A-S due to end of useful life
- December 9, 2009, Change setpoints and adjust limit stop setting on containment vacuum relief differential pressure switch CVRIDPIS5220A

The inspectors selected these activities based upon the structure, system, or component's ability to affect risk. 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 specifications, 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 postmaintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of seven postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the Unit 3 refueling outage, conducted October 19, 2009, through December 4, 2009, 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 cooldown processes and monitored licensee controls over the outage activities listed below.

- Configuration management, including maintenance of defense-in-depth, is 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 primary containment 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

- Review of Operating Experience Smart Sample FY2007-03, crane and heavy lift inspection

- Review of Operating Experience Smart Sample FY2007-01, related to Information Notice 2006-20

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

These activities constitute completion of one refueling outage and other outage inspection sample as defined in Inspection Procedure 71111.20-05.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, procedure requirements, and technical specifications to ensure that the two surveillance activities listed below demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions. The inspectors either witnessed 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.

- November 9, 2009, Train B integrated emergency diesel generator/engineering safety features test (Operating Procedure OP-903-116)
- November 18, 2009, Leak test on reactor coolant loop 1 shutdown cooling outside containment isolation valve SI-407B
- December 14, 2009, Annulus negative pressure valves ANP-101 and ANP-102 surveillance test (Operating Procedure OP-903-120)

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

These activities constitute completion of three surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

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

.1 Inoffice Review, Revision 23

a. Inspection Scope

The inspector performed an in-office review of Emergency Plan Implementing Procedure EP-001-001, Revision 23, "Recognition and Classification of Emergency Conditions," submitted August 19, 2009. This revision

- Added information to emergency action level CU1 to clarify that steam generator leakage is considered to be identified reactor coolant leakage

- Added information to emergency action level RCB2 to clarify that manual initiation of emergency core cooling systems to compensate for a steam generator tube leak/rupture meets the intent of the emergency action level

- Added information to emergency action level HU6 to clarify that entry conditions are not met until hurricane force winds are projected for the site occurring in less than or equal to twelve hours

This revision was compared to its previous revision, 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 Energy Institute Report 99-01, "Emergency Action Level Methodology," Revision 5, and to the standards in 10 CFR 50.47(b) to determine if the revision adequately implemented the requirements of 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection.

These activities constitute completion of one sample as defined in Inspection Procedure 71114.04-05.

b. Findings

No findings of significance were identified.

.2 Inoffice Review, Revision 24

a. Inspection Scope

The inspector performed an in-office review of the Waterford Steam Electric Station Emergency Plan, Revision 38, and Emergency Plan Implementing Procedure EP-001-001, "Recognition and Classification of Emergency Conditions," Revision 24, submitted October 23, 2009. These revisions

- Deleted emergency action level CU4, fuel clad degradation

- Changed the initiating conditions of Emergency Action Level SU9, Fuel Clad Degradation, from greater than 1.0 μCi/g DEI or greater than 100 over E-Bar μCi/g, to greater than 60 μCi/g DEI or greater than 1.0 μCi/g DEI for more than a continuous 48 hour period or greater than 100 over E-Bar μCi/g

- Removed fuel clad degradation from the list of Unusual Event conditions on the Emergency Plan Table 4-1, "Summary of Initiating Conditions," and the index of initiating conditions for cold shutdown conditions in Procedure EP-001-001

The NRC approved the licensee's changes to emergency action levels CU4 and SU9 in a Safety Evaluation Report and letter dated October 13, 2009 (Agency Document and Management System Accession Number ML092600263).

These revisions were compared to the Safety Evaluation Report dated October 13, 2009, to determine if the revisions adequately implemented the requirements of 10 CFR 50.54(q).

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

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Training Observations

a. Inspection Scope

The inspectors observed a training evolution for licensed operators on December 21, 2009, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator data regarding drill and exercise performance. The inspectors reviewed the event scenarios and crew briefings for two scenarios. The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the postevolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the corrective action program.

These activities constitute completion of one sample as defined in Inspection Procedure 71114.06-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess licensee personnel's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. 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 independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone
- Controls (surveys, posting, and barricades) of radiation, high radiation, or airborne radioactivity areas
- Radiation work permits, procedures, engineering controls, and air sampler locations
- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms
- Barrier integrity and performance of engineering controls in airborne radioactivity areas
- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools
- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls
- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions
- Adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination control during job performance
- Dosimetry placement in high radiation work areas with significant dose rate gradients
- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas
- Controls for special areas that have the potential to become very high radiation areas during certain plant operations
- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas
- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

Either because the conditions did not exist or an event had not occurred, no opportunities were available to review the following items:

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent

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

These activities constitute completion of 21 of the required 21 samples as defined in Inspection Procedure 71121.01-05.

b. Findings

Introduction. The inspectors reviewed a Green self-revealing, noncited violation of Technical Specification 6.8.1 which resulted from a worker failing to follow radiation protection procedures.

Description. On November 17, 2009, a contract radiation worker went to work near steam generator 1 and received multiple electronic dosimeter dose rate alarms, but did not leave the area until receiving a continuous dose alarm. In response, the licensee investigated and found the worker indicated to radiation protection access control personnel he would be going to the D-ring to work. The radiation protection technician providing the radiological briefing showed the worker a map of reactor coolant pump 1A and asked if that was where individual would be working. The worker acknowledged it was, and the radiation protection technician used the survey map associated with Radiation Work Permit 618, Task 1, "Remove/Replace Insulation in the Reactor

Containment Building," to brief the worker on the radiological conditions. The worker then signed onto Radiation Work Permit 618, Task 1, which provided a dose alarm setpoint of 50 millirem and dose rate setpoint of 350 millirem per hour, and went to work near steam generator 1, where dose rates were higher than the area for which the worker was briefed. The licensee determined the worker entered a maximum dose rate of 763 millirem per hour and received a dose of 50.8 millirem. Radiation protection representatives stated the appropriate radiation work permit for the work area was Radiation Work Permit 618, Task 2. Through examination of the electronic dosimeter histogram, the licensee verified the worker received multiple dose rate alarms. The worker mistakenly thought the dose rate alarms were generated by the worker's powered air purifying respirator signaling low air flow. Additional corrective actions were being considered at the time of the inspection.

Analysis. The failure to follow radiation protection procedural requirements for entry into the radiological controlled area was a performance deficiency. This finding is greater than minor because it involved the program attribute of exposure control and affected the cornerstone objective in that the failure of the worker to follow procedural guidance resulted in the worker being unknowledgeable of the dose rates in all areas entered. The inspectors used the Occupational Radiation Safety Significance Determination Process and determined the finding had very low safety significance because it was not: (1) an as low as reasonably achievable (ALARA) finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an inability to assess dose. The finding had a crosscutting aspect in the area of human performance, work practices component, because the worker failed to use human error prevention techniques such as self and peer checking [H.4.a].

Enforcement. Technical Specification 6.8.1 requires written procedures be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Appendix A lists procedures for access control to radiation areas. Procedure EN-RP-100, "Radworker Expectations," Revision 3, Section 5.3[9], requires the radiation work permit to be read, understood, and obeyed as a condition of radiologically controlled area access. Section 5.4[3](h) requires the worker know where to properly perform his/her task. Section 5.3[17] requires the worker be briefed and sign on the appropriate radiation work permit. Section 5.3[11] requires the worker know the radiological conditions in the work area. The contract worker violated these requirements when the worker did not know where to perform his/her task, did not sign the appropriate radiation work permit and task, and did not know the radiological conditions in the work area as evidenced by the multiple electronic dosimeter dose rate alarms. Because this failure to follow radiation protection procedural guidance when entering the radiological controlled area was of very low safety significance and has been entered into the licensee's corrective action program in Condition Reports WF3-2009-05648 and WF3-2009-06852, this violation is being treated as an noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/2009005-01; "Failure to Follow Radiation Protection Procedural Requirements."

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors assessed licensee personnel's performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable. The inspectors used the requirements in 10 CFR Part 20 and the licensee's procedures required by technical specifications as criteria for determining compliance. The inspectors interviewed licensee personnel and reviewed the following:

- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents

- Shielding requests and dose/benefit analyses

- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding

- Workers' use of the low dose waiting areas

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

- Corrective action documents related to the ALARA program and follow-up activities, such as initial problem identification, characterization, and tracking

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

These activities constitute completion of two of the required 15 samples and four of the optional samples as defined in Inspection Procedure 71121.02-05.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope
The inspectors performed a review of the data submitted by the licensee for the third quarter 2009 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 of significance were identified.

.2 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system specific activity performance indicator for the period from the third quarter 2008 through the third quarter 2009. 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 5. The inspectors reviewed the licensee's reactor coolant system chemistry samples, technical specification requirements, issue reports, event reports, and NRC integrated inspection reports for the period of the third quarter 2008 through the third quarter 2009 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. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a reactor coolant system sample. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one reactor coolant system specific activity sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.3 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system leakage performance indicator for the period from the third quarter 2008 through the third quarter 2009. 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 5. The inspectors reviewed the licensee's operator logs, reactor coolant system leakage tracking data, issue reports, event reports, and NRC integrated inspection reports for the period of the third quarter 2008 through the third quarter 2009 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 reactor coolant system leakage sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.16 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences performance indicator for the third quarter 2009. 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 5, was used. The inspectors reviewed the licensee’s assessment of the performance indicator for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee’s performance indicator data collection and analyses, the inspectors discussed with radiation protection staff, the scope and breadth of its data review, and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas.

These activities constitute completion of the occupational radiological occurrences sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences performance indicator for the third quarter 2009. 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 5, was used. The inspectors reviewed the licensee’s issue report database and selected individual reports generated since this indicator was last reviewed to identify any 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 of significance 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 Review 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. 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.

c. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. The inspectors accomplished this through review of the station's daily corrective action documents.

The inspectors performed these daily reviews as part of their daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

.3 Selected Issue Follow-up Inspection

a. Inspection Scope

During a review of items entered in the licensee's corrective action program, the inspectors reviewed conditions surrounding reactor coolant system leakage and boric acid corrosion related to reactor coolant pumps. The inspectors considered the following during the review of the licensee's actions: (1) complete and accurate identification of problems in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

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

b. Findings

i. Introduction. A self-revealing Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, was identified for the licensee's failure to promptly correct a condition adverse to quality. Specifically, the licensee did not promptly correct reactor coolant pump vapor seal leakage that resulted in boric acid accumulation on the component cooling water heat exchanger and cover areas of three reactor coolant pumps. Corrective actions for this condition were implemented during Refueling Outage 15, but these corrective actions failed to correct the condition and the vapor seal leakage continued through Operating Cycle 16. This resulted in some additional boric acid corrosion and degradation to reactor coolant pump covers and carbon steel component cooling water flanges.

Description. The reactor coolant pumps are designed to direct vapor stage seal leakage to the reactor drain tank via installed piping which includes a check valve to prevent back flow from the drain line to the vapor seal. For several cycles, the licensee has recognized that vapor stage seal leakage has not been draining to the reactor drain tank as designed but has instead been backing up in the line and spilling into the pump shroud region. It was theorized that this failure of the vapor stage leakage to flow to the reactor drain tank was due to the normally positive pressure in the reactor drain tank and that a design change was needed. During Refueling Outage 15, the licensee implemented Engineering Change EC-6256 to redirect all reactor coolant pump vapor seal leakage flow to a floor drain instead of the reactor drain tank. However, the design change did not consider the flow restriction effects of an existing check valve in each of the reactor coolant pump vapor stage leakage piping, and made the modification downstream of each of those existing check valves such that vapor stage leakage no longer faced the back pressure from the reactor drain tank, but still had to pass through the existing check valves in order to reach the target floor drain.

The postmaintenance test prescribed by Engineering Change EC-6256 to verify flow through the modified vapor stage leakage piping from the seal, through the leak-off piping (including the installed check valve) to the floor drain was not implemented as specified. Instead, because of schedule and resource impacts (it would have been difficult, resource intensive, and intrusive to conduct the test as prescribed), a substitute postmaintenance test was performed that only verified flow through the portion of the piping that was modified. This meant that the postmaintenance test did not verify that water would actually flow from the vapor stage seal, through the existing check valves, through the new piping modification and into the floor drain.

Operating Cycle 16 proceeded following Refueling Outage 15 with the newly modified and inadequately tested vapor stage leakage line in operation. At the conclusion of Operating Cycle 16, Mode 3 walkdowns at the beginning of Refueling Outage 16 identified more boric acid accumulation on three of four reactor coolant pumps, indicating continued reactor coolant pump vapor stage leakage out onto the heat exchanger and pump cover. The licensee's root cause analysis determined that Engineering Change EC-6256 was ineffective. A test similar to the postmaintenance test originally prescribed by Engineering Change EC-6256 was performed on reactor coolant pump 2B (which had experienced the most boric acid accumulation) and it identified that the installed check valve RC-511B was incapable of passing flow as intended by design. The valve was a 3/4" Velan spring loaded check valve in which the pressure required to overcome the spring load was more than the static head of water between the vapor stage seal and the check valve could develop. Both the original design and the subsequent design modification implemented by Engineering Change EC-6256 were incapable of passing flow as intended by design because the vapor stage leakage line between the seal and the check valve could not develop enough static head to lift the check valve before backing up and spilling over onto the pump heat exchanger and cover. If the postmaintenance test prescribed by Engineering Change EC-6256 had been implemented as prescribed during Refueling Outage 15, this design flaw associated with the check valve would have been detected and the design could have been modified to correct this condition at that time. However, because that postmaintenance test was not properly implemented, the condition adverse to quality (the vapor stage leakage onto the reactor coolant pump heat exchanger and pump cover and associated boric acid accumulation and associated corrosion) continued to exist for another operating cycle.

Analysis. The licensee's failure to promptly correct a condition adverse to quality is a performance deficiency. The finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and affects the cornerstone objective to limit the likelihood of those events that upset plant stability. Using the Manual Chapter 0609, Attachment 4, Phase 1 screening worksheet, the issue screened as having very low safety significance because, although the finding contributes to the likelihood of a reactor trip, mitigation equipment is still available. This finding had a crosscutting aspect in the area of human performance associated with work control in that the licensee did not effectively plan for the resources necessary to implement the postmaintenance testing per Engineering Change EC 6256 [H.3(a)].

Enforcement. Title10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, the licensee failed to promptly correct a condition adverse to quality. Specifically, the licensee failed to correct the reactor coolant pump vapor seal leakage with the corrective actions it implemented during Refueling Outage 15 (ending May 31, 2008), and the vapor seal leakage continued through operating cycle 16 until corrected during Refueling Outage 16 (ending December 4, 2009). Because this finding was of very low safety significance and has been entered into the licensee's corrective action program as Condition Report CR-WF3-2009-5501, it is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000382/2009005-02, "Reactor Coolant Pump Vapor Seal Leakage."

ii. Introduction: A self-revealing Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, was identified for the licensee's failure to prescribe an activity affecting quality by documented instructions, procedures, or drawings appropriate to the circumstance. Specifically, for all reactor coolant pump heat exchanger to pump cover bolted connection gasket replacements between the refueling outage of 1986 (Refueling Outage 1) and the refueling outage of 2009 (Refueling Outage 16), the licensee prescribed the wrong gasket material, gasket size, and fastener preload because they had failed to incorporate a design change implemented during Refueling Outage 1 into their instructions, procedures, or drawings. Station Modification Package SMP-1427, an engineering change implemented during Refueling Outage 1 in response to industry operating experience, called for a thicker gasket, different gasket material, and an increased bolt preload in order to increase gasket compression and reduce the probability of leakage. As a consequence of failing to incorporate Station Modification Package SMP-1427 changes into procedures, all heat exchanger gasket replacements since Refueling Outage 1, four gasket replacements in total, have utilized thinner gaskets with less than the vendor recommended compression.

Description. After the licensee's first operating cycle, industry operating experience indicated that the reactor coolant pump heat exchanger to pump cover bolted connection had a high probability of leakage as designed and warranted a design modification to increase gasket compression to reduce the likelihood of reactor coolant leakage at that interface. As a result, the licensee implemented a design modification, Station Modification Package SMP-1427, to change the required gasket material from stainless steel/asbestos to inconel/grafoil, to change the gasket thickness from 0.125 inches to 0.135 inches, and to change the fastening method from 2200-foot pounds of torque (roughly equivalent to 30 ksi tensioned) to 38.7 ksi tensioned.

All four reactor coolant pump bolted connections were modified to the new gaskets and fastening method as prescribed in Station Modification Package SMP-1427. However, Technical document TD-B580.0025 was not updated with the design change at that time. As a result, all gasket replacements conducted between Refueling Outage 1 and Refueling Outage 16 were accomplished in accordance with the outdated and inadequate specifications that remained in TD-B580.0025. The result was that, by the beginning of Refueling Outage 16, only reactor coolant pump RCP-1B still retained the modifications prescribed by Station Modification Package SMP-1427 and implemented in Refueling Outage 1.

It is noteworthy that the inspection of reactor coolant pump 1A during the midcycle outage on October 9, 2007, identified a sizable quantity of boric acid crystals contained in the pump shroud. The root cause analysis concluded that the boric acid accumulation was primarily due to leakage past the reactor coolant pump heat exchanger to pump cover gasket. However, the root cause analysis for this leakage did not identify that operating experience associated with leakage past these gaskets had caused the licensee to implement Station Modification Package SMP-1427 in Refueling Outage 1, and neither did the root cause analysis identify that the thicker gasket and modified fastening method were needed to achieve the vendor's recommended compression. Therefore, the gasket replacement on reactor coolant pump RCP-1A was not performed in accordance with Station Modification Package SMP-1427. In addition, it is noteworthy that boric acid accumulation discovered on reactor coolant pump RCP-2B on October 20, 2009, prompted another root cause analysis by the licensee which concluded that leakage past the heat exchanger to pump cover gasket may have been a possible cause of a portion of that boric acid accumulation. The root cause analysis performed in 2007 for reactor coolant pump RCP-1A was a missed opportunity to identify the licensee's past failure to include the Station Modification Package SMP-1427 design modifications into plant procedures. Had that opportunity not been missed, it is postulated that the inadequate gasket and fastener configuration on reactor coolant pump RCP-2B may have been identified and corrected before the discovery of significant boric acid accumulation on it during Operating Cycle 16, which may have reduced the accumulation of boric acid on that pump.

Analysis. The licensee's failure to prescribe appropriate gasket replacement requirements in instructions, procedures, or drawings is a performance deficiency. The finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and affects the cornerstone objective to limit the likelihood of those events that upset plant stability. Using the Manual Chapter 0609, Attachment 4, Phase 1 screening worksheet, the issue screened as having very low safety significance because, although the finding contributes to the likelihood of a reactor trip, mitigation equipment is still available. This finding had a crosscutting aspect in the area of problem identification and resolution associated with operating experience in that the licensee did not institutionalize operating experience through changes to the station procedures [P.2(b)].

Enforcement. Title10 CFR Part 50, Appendix B, Criterion V, requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Contrary to the above, the licensee failed to prescribe an activity affecting quality by instructions, procedures, or drawings, of a type appropriate to the circumstances. Specifically, for all reactor coolant pump heat exchanger to pump cover bolted connection gasket replacements between the refueling outage of 1986 (Refueling Outage 1) and the refueling outage of 2009 (Refueling Outage 16), the licensee prescribed the wrong gasket material, gasket size, and fastener preload because they had failed to incorporate a design change implemented during Refueling Outage 1 into their instructions, procedures, or drawings. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program as Condition Report CR-WF3-2009-5501, it is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000382/2009005-03, "Failure to Update Drawings after Design Change."

4OA5 Other Activities

.1 Temporary Instruction 2515-172, "Reactor Coolant System Dissimilar Metal Butt Welds"

a. Inspection Scope

The reactor coolant system for this unit is carbon steel with stainless steel cladding and has the following dissimilar metal welds subject to the requirements of the Materials Reliability Program 139:

1. One 12-inch pressurizer surge line nozzle was mitigated during a previous outage using a weld overlay process. The weld was classified as Category F per materials reliability program guidelines.

2. Three 6-inch pressurizer safety nozzles were mitigated during a previous outage using a weld overlay process. The welds were classified as Category F per materials reliability program guidelines.

3. One 4-inch pressurizer spray nozzle was mitigated during a previous outage using a weld overlay process. The weld was classified as Category F per materials reliability program guidelines.

4. Two 14-inch hot leg shutdown cooling nozzles were mitigated during a previous outage using a weld overlay process. The welds were classified as Category F per materials reliability program guidelines.

5. One 12-inch hot leg surge nozzle was mitigated during a previous outage using a weld overlay process. The weld was classified as Category F per materials reliability program guidelines.

6. One 2-inch hot leg drain nozzle was mitigated during a previous outage using a weld overlay process. The weld was classified as Category F per materials reliability program guidelines.

7. Four 12-inch safety injection nozzles were previously left unmitigated. The licensee performed a volumetric inspection of each nozzle during the current outage and classified the welds as Category E per materials reliability program guidelines.

8. Four 30-inch reactor coolant pump suction piping (unmitigated as of this outage). The licensee performed a volumetric inspection of each pipe during the current outage and classified the welds as Category E per materials reliability program guidelines.

9. Four 30-inch reactor coolant pump discharge piping (unmitigated as of this outage). The licensee performed a volumetric inspection of each pipe during the current outage and classified the welds as Category E per materials reliability program guidelines.

All of the pressurizer and hot-leg welds have been mitigated, in previous outages, using a full-structural overlay weld. The cold-leg-temperature welds have not been mitigated as of this outage. The cold-leg welds have been volumetrically inspected and any decision to mitigate these welds will be made on the basis of these and/or future inspections.

03.01 <u>Licensee's Implementation of the Materials Reliability Program (MRP-139) Baseline Inspections</u>

a. The inspector reviewed records of structural weld overlays and nondestructive examination activities associated with the licensee's hot leg surge nozzle's structural weld overlay mitigation effort.

b. The licensee was not planning to take any deviations from the baseline inspection requirements of Materials Reliability Program MRP-139, and all other applicable dissimilar metal butt welds were scheduled in accordance with Materials Reliability Program MRP-139 guidelines.

03.02 Volumetric Examinations

a. The inspector observed the phased array ultrasonic examination of two cold leg welds that were not scheduled to be overlaid. This examination was conducted in accordance with ASME Code, Section XI, Supplement VIII Performance Demonstration Initiative requirements regarding personnel, procedures, and equipment qualifications. No relevant conditions were identified during this examination.

b. The inspector reviewed records for the nondestructive evaluations performed on the hot leg surge nozzle weld overlay. Inspection coverage met the requirements of Materials Reliability Program MRP-139 and no relevant conditions were identified.

c. The certification records of ultrasonic examination personnel were reviewed for those personnel that performed the examinations of the cold-leg welds. All personnel records showed that they were qualified under the EPRI Performance Demonstration Initiative.

d. No deficiencies were identified during the nondestructive examinations.

03.03 Weld Overlays

a. The inspector reviewed the welding activities associated with the weld overlay performed on the hot leg surge nozzle.

b. The licensee submitted and received NRC authorization for the use of relief request from the ASME code to apply weld overlays on their dissimilar metal butt welds. Using this, the licensee performed weld overlays on all of the dissimilar metal butt welds associated with pressurizer and hot leg temperatures. This welding took place in previous outages. The inspector reviewed the weld records for one of these welds to ensure the welding was performed in accordance with the ASME code as modified by the approved relief requests.

c. No deficiencies were identified in the completed full structural weld overlays.

03.04 Mechanical Stress Improvement

This item was not applicable because the licensee did not have plans to employ a mechanical stress improvement process.

03.05 Inservice Inspection Program

The inspector reviewed the licensee's risk informed inservice plan and verified that all dissimilar metal butt welds have been entered into the plan and will be examined on a schedule consistent with Materials Reliability Program MRP-139.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

On October 1, 2009, the inspector conducted a telephonic exit meeting to present the results of the in-office inspection of changes to the Waterford Steam Electric Station, Unit 3's, emergency action levels to Mr. J. Lewis, Manager, Emergency Preparedness. He acknowledged the issues presented. The inspector asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified

On November 9, 2009, the inspector conducted a telephonic exit meeting to present the results of the in-office inspection of changes to the Waterford Steam Electric Station, Unit 3', emergency plan and emergency action levels to Mr. R. Perry, Acting Emergency Preparedness Manager. He acknowledged the issues presented. The inspector asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On November 13, 2009, the inspectors presented the results of the inservice inspection to you and other members of your staff. You acknowledged the issues presented. The inspectors returned proprietary material examined during the inspection.

On November 20, 2009, the inspectors presented the inspection results to Mr. C. Arnone, General Manager, Plant Operations, and other members of your staff. They acknowledged the issues presented. The inspector asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On January 11, 2010, the inspectors presented the quarterly inspection results to you and other members of your staff. You acknowledged the issues presented. The inspectors asked 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 VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as noncited violations.

Technical Specification 6.8.1 requires written procedures be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Appendix A lists procedures for access control to radiation areas. Procedure EN-RP-100, "Radworker Expectations," Revision 3, Section 5.3[9] requires the radiation work permit to be read, understood, and obeyed as a condition of radiologically controlled area access. Procedure EN-RP-100, "Radworker Expectations," Revision 3, Section 5.4[3](h) requires the worker know where to properly perform his/her task. Section 5.3[17] requires the worker be briefed and sign on the appropriate radiation work permit. Section 5.3[11] requires the worker know the radiological conditions in the work area. The licensee identified an example of a worker entering a high radiation area using an inappropriate radiation work permit and without knowing the dose rates in the area. On October 24, 2009, a security officer entered shutdown heat exchanger Room B and received an electronic dosimeter dose rate alarm. The room was posted as a high radiation area and dose rates within the area were as high as 140 millirem per hour. The officer entered the radiological controlled area using Radiation Work Permit 2009005, "Tours and Inspection in All Radiological Controlled Areas, Except High Radiation Areas, Locked High Radiation Areas, Very High Radiation Areas, and the Reactor Containment Building." Because the radiation work permit did not allow entry into high radiation areas, radiation protection personnel did not anticipate the officer would enter the room and did not brief the officer on the dose rates in the area. In response, the licensee conducted a human performance error review and counseled the officer. This finding was of very low safety significance because it did not involve an actual or substantial potential of an overexposure. This finding was entered into the licensee's corrective action program as Condition Report CR-WF3-2009-05648.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

C. Arnone, General Manager Plant Operations
D. Bauman, Senior Project Manager
M. Bratton, Manager, Senior Lead Technical Specialist
J. Brawley, ALARA Supervisor, Radiation Protection
B. Celeste, Lead Level III, Contractor, C&S Engineers, Inc.
K. Cook, Acting General Manager Plant Operations
L. Dauzat, Supervisor, Radiation Protection
D. Dufrene; Technician, Radiation Protection
G. Ferguson, PE, IWE Examination
J. Gobell, Project Manager
J. Houghtaling, Senior Project Manager
C. Hunsaker, Technical Specialist II
J. Kowalewski, Vice President, Operations
J. Lewis, Manager, Emergency Preparedness
R. Luter, Technical Specialist IV
M. Mason, Engineer, Licensing
R. McGaha, Technical Specialist II
M. Mason, Engineer, Licensing
R. Murillo, Manager, Licensing
K. Nichols, Director, Engineering
R. O'Quinn, Senior Staff Engineer
C. Pickering, Supervisor, Mechanical Maintenance
B. Piluti, Manager, Radiation Protection
J. Polluck, Engineer, Licensing
R. Redmond, Technical Specialist, Boric Acid Corrosion Control Program
W. Sims, Manager, Major Projects I
B. Williams, Technical Specialist IV
R. Williams, ASME Section XI/ISI Senior Lead

NRC Personnel

M. Haire, Senior Resident Inspector
D. Overland, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000382/2009005-01	NCV	Failure to follow radiation protection procedural requirements
05000382/2009005-02	NCV	Reactor Coolant Pump Vapor Seal Leakage

Opened and Closed

05000382/2009005-01	NCV	Failure to follow radiation protection procedural requirements
05000382/2009005-02	NCV	Reactor Coolant Pump Vapor Seal Leakage
05000382/2009005-03	NCV	Failure to Update Drawings after Design Change

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
WSES-FSAR-UNIT-3	Final Safety Analysis Report – Section 2.4, Hydrologic Engineering	10
OP-901-521	Off-Normal Procedure for Severe Weather and Flooding	301

Section 1R04: Equipment Alignment

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
OP-002-004	Chilled Water System	303
OP-903-063	Chilled Water Pump Operability Verification	302
SD-CHW	Essential Chilled Water System Description	6
G853 Sheet 3	Chilled Water flow Diagram SH-1	December 4, 1975
SD-SI	Safety Injection System Description	13
OP-009-008	Safety Injection System Operating Procedure	26

Section 1R05: Fire Protection

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
OP-009-004	Fire Protection	305
MM-004-424	Building Fire Hose Station Inspection and Hose Replacement	10
MM-007-010	Fire Extinguisher Inspection and Extinguisher Replacement	302
FP-001-014	Duties of a Fire Watch	14
FP-001-015	Fire Protection Impairments	302
DBD-018	Appendix R/Fire Protection	
FP-001-015	Fire Protection Impairments	302
FP-001-018	Pre-fire Plan Strategies, Development, And Revision	300
UNT-007-006	Housekeeping	301
EN-DC-161	Control of Combustibles	003
UNT-007-060	Control of Loose Items	302
UNT-005-013	Fire Protection Program	010
SD-FP	Fire Protection System Description	2

Section 1R06: Flood Protection Measures

CONDITION REPORTS

CR-WF3-2005-03338 CR-WF3-1996-00930 CR-WF3-2009-3925

PROCEDURE/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
WSES-FSAR-UNIT-3	Appendix 3.6A Pipe Rupture Analysis	February 2002
WSES-FSAR-UNIT-3	Water Level (Flood) Design	February 2002
WSES-FSAR-UNIT-3	System Description Plant Sumps	6
OP-901-521	Severe Weather and Flooding	301
G-349	Yard Duct Runs and Outdoor Lighting Drawing	18

Section 1R07: Heat Sink Performance

PROCEDURES/DOCUMENTS

NUMBER	TITLE	DATE
NOECP-257	Steam Generator Secondary Side Inspections	4
LTR-SGDA-08-129	Acceptability of Loose Batwing Section found in the Upper Central Stay Cavity Region during RF15	May 12, 2008
LTR-SGDA-09-189	Acceptability of SG Operation As a Result of an Unattached Steam Vent and Observed Feedwater Ring Erosion	November 16, 2009
LTR-SGDA-09-188	Acceptance Criteria for Waterford Feedwater Discharge Elbows	November 13, 2009

Section 1RO8: Inservice Inspection Activities

DOCUMENTS/PROCEDURES/REPORTS

NUMBER	TITLE	REVISION / DATE
EN-DC-317	Entergy Steam Generator Administrative Procedure	4

DOCUMENTS/PROCEDURES/REPORTS

NOECP-257	Steam Generator Secondary Side Inspection	4
NOECP-252	Steam Generator Eddy Current Inspection Testing	11
CEP-NDE-0955	Alloy 600 Visual Examination (VE) of Bare-Metal Surfaces	301
EN-DC-319	Inspection and Evaluation of Boric Acid Leaks	4
NOECP-107	Boric Acid corrosion Control Program	3
WF3-CHEM-SEC-001-06	Strategic Secondary Water Chemistry Plan	6
WDI-PJF-1304321-FSR-001	Waterford 3 – RF16 – Reactor Vessel Head Penetration Inspection Final Report.	0
WDI-SSP-1002	Reactor Vessel Head Penetration Inspection Tool Operation for ANO 2 and Waterford 3 – ROSA	3
WCAL-002	Pulser/Receiver Linearity Procedure	10
WDI-ET-003	IntraSpect Eddy Current Imaging Procedure for Inspection of Reactor Vessel Head Penetrations	14
WDI-ET-004	IntraSpect Eddy Current Analysis Guidelines	14

WDI-STD-1040	IntraSpect Ultrasonic Procedure for Inspection of Reactor Vessel Head Penetrations, Time of Flight Ultrasonic, Longitudinal Wave and Shear Wave	2
WDI-STD-1041	IntraSpect UT Analysis Guidelines	1
WDI-STD-101	RVHI Vent Tube J-Weld Eddy Current Examination	8

<u>DOCUMENTS/PROCEDURES/REPORTS</u>

WDI-STD-114	RVHI Vent Tube ID & CS Wastage Eddy Current Examination	10
CEP-NDE-0404	Manual Ultrasonic Examination of Ferritic Piping Welds (ASME XI)	4
ISI-UT-09-019	UT Calibration/Examination (WO 157687) – RCS Cold Leg Loop 1A – Weld No. 07-005	October 31, 2009
L-09-006	Ultrasonic Instrument Linearity – Krautkramer USN 60 SW (Serial No. 01VNCT); Transducer Frequency 4.0 MHz (Serial No. 5746222529); Calibration Standard (Serial No. 9634); Couplant – Ultragel II (Batch No. 06225)	October 22, 2009
ISI-VT-09-194	Visual Examination for Boric Acid Detection (WO 159119) – RCS Loop 1A Cold Leg – Weld No. 07-002	October 27, 2009
MRP-139	Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline	1
CEP-NDE-0901	VT-1 Examination	4
CEP-NDE-0902	VT-2 Examination	7
CEP-NDE-0903	VT-3 Examination	5
SI-UT-130	Procedure for the Phased Array Ultrasonic Examination of Dissimilar Metal Welds	3

SI-NDE-06	Calibration of Ultrasonic NDE Equipment	4
SI-NDE-08	Qualification and Certification of NDE Personnel for Nuclear Applications	1
WF3 11-002 RCP 2A Suction Nozzle	Structural Integrity Associates - Phased Array Ultrasonic Examination Record Data Sheet for Weld No. 11-002: Reactor Coolant Pump 2A Cold Leg Suction Nozzle	October 30, 2009
WF3 11-002 RCP 2A Suction AX SH	Structural Integrity Associates – Ultrasonic Phased Array Calibration Record for Weld No. 11-002: Reactor Coolant Pump Suction Nozzle Dissimilar Metal Weld – Wedge Angle 36.2° (Axial Scan)	October 30, 2009
WF3 11-002 RCP 2A Suction Circ – 10 RL	Structural Integrity Associates – Ultrasonic Phased Array Calibration Record for Weld No. 11-002: Reactor Coolant Pump Suction Nozzle Dissimilar Metal Weld – Wedge Angle 4.0° (Circumferential Scan)	October 30, 2009
WF3 11-002 RCP 2A Suction Circ + 10 RL	Structural Integrity Associates – Ultrasonic Phased Array Calibration Record for Weld No. 11-002: Reactor Coolant Pump Suction Nozzle Dissimilar Metal Weld – Wedge Angle 14.0° (Circumferential Scan)	October 30, 2009
WF3 11-002 RCP 2A Suction Flat RL	Structural Integrity Associates – Ultrasonic Phased Array Calibration Record for Weld No. 11-002: Reactor Coolant Pump Suction Nozzle Dissimilar Metal Weld – Wedge Angle 14.0° (Axial & Circumferential Scan)	October 30, 2009
WF3-LIN-09-002	Structural Integrity Associates – Ultrasonic Linearity Record – Zetec/RD Tech OmniScan MX – Version 1.4R3 (Serial No. ONMI-1983); Transducer 115-000-613 (Serial No. 01VTVW); Reference Block 16" AX (Serial No. SI-16-AX-03).	October 21, 2009
Product Code 115-000-566	Krautkramer Phased Array Transducer Certificate of Compliance (Serial No. 01VM4k-1)	September 02, 2008

SII006-07-09-28155-1	Laboratory Testing Inc. – Certified Test Report for Sonotech Ultragel II	July 27, 2007
WF3 12-009 RCP 2A Safety Injection Nozzle	Structural Integrity Associates - Phased Array Ultrasonic Examination Record Data Sheet for Weld No. 12-009: RCP 2A Safety Injection Nozzle	October 29, 2009
WF3 12-009 RCP 2A Safety Injection AX SH	Structural Integrity Associates – Ultrasonic Phased Array Calibration Record for Weld No. 12-009: Reactor Coolant Pump 2A Safety Injection Nozzle Dissimilar Metal Weld – Wedge Angle 36.2^{o} (Axial Scan)	October 29, 2009
WF3 12-009 RCP 2A Safety Injection AX RL	Structural Integrity Associates – Ultrasonic Phased Array Calibration Record for Weld No. 12-009: Reactor Coolant Pump 2A Safety Injection Nozzle Dissimilar Metal Weld – Wedge Angle 16.2^{o} (Axial Scan)	October 29, 2009
WF3 12-009 RCP 2A Safety Injection CIRC RL	Structural Integrity Associates – Ultrasonic Phased Array Calibration Record for Weld No. 12-009: Reactor Coolant Pump 2A Safety Injection Nozzle Dissimilar Metal Weld – Wedge Angle 16.2^{o} (Circumferential Scan)	October 29, 2009
Contract No. C-08-422	Sonaspection – Structural Integrity of Calibration Block No. SI-16-AX-03 & SI-16-CIRC-03	December 17, 2009
WF3-LIN-09-003	Structural Integrity Associates – Ultrasonic Linearity Record – Zetec/RD Tech OmniScan MX – Version 1.4R3 (Serial No. ONMI-1590); Transducer 115-000-613 (Serial No. 01VTW0); Reference Block 16” AX (Serial No. SI-16-AX-03).	October 21, 2009
Product Code 115-000-613	Krautkramer Phased Array Transducer Certificate of Conformity (Serial No. 01VTW0-1)	August 26, 2008

ENGINEERING CHANGE REQUEST

NUMBER	TITLE	DATE
0000004490	Steam Generator Degradation Assessment and Repair Criteria for RF15	April 2008
0000005544	Waterford 3 Cycle 16 Steam Generator Operational Assessment	April 2008
0000005544	Waterford 3 Cycle 16 Steam Generator Operational Assessment	August 2008
0000008593	Waterford-3 RF16 Steam Generator Eddy Current Probe Equivalency Report	Revision 0
0000008594	Waterford-3 RF16 Steam Generator Inspection ECT Data Analyst Training Manual	
0000008592	RF16 Waterford-3 Steam Generator Analysis Guidelines	Revision 0
0000008591	Steam Generator Degradation Assessment and Repair Criteria for RF16	October 2009

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
ECR-WF3-4490	Steam Generator Degradation Assessment and Repair Criteria	April 2008
W3F1-2008-0039	Steam Generator Conditions Observed at Waterford 3 During Refueling Outage 15	May 20, 2008
ECR-WF3-8593	Waterford -3 RF16 Steam Generator Eddy Current Probe Equivalency Report	November 3, 2009
ECR-WF3-8594	Document the Analysts Training Manual for RF16 SG Eddy Current Analysts per the Requirements of NEI 97-06 and EN-DC-317	November 6, 2009
ECR-WF3-8592	RF16 Waterford-3 Steam Generator Analysis Guidelines	November 5, 2009
ECR-WF3-8591	Steam Generator Degradation Assessment and Repair Criteria for RF16	October 2009

WF3-CHEM-SEC-001-06	Strategic Secondary Water Chemistry Plan	6
	Inspection Report for Bare Metal Visual of Reactor Vessel Head	
BOP-VT-09-020	Visual Examination of Boric Acid Detection	November 12, 2009
LTR-SGMP-09-179	Estimate of Through-Tube Depth of Intrados Wear Scar in Waterford Steam Generator 32	November 10, 2009
LO-WLO-2008-00068	WF3 Boric Acid corrosion Control Program Self-Assessment	October 6-16, 2008
LO-WLO-2006-00046	Waterford 3 Strategic Secondary Water Chemistry Plan Self-Assessment	March 27-30, 2006
LO-WLO-2008-0091	Benchmark of: Point Beach (PBNP) Nuclear Plant	July 16-17, 2009
W3F1-2008-0039	Steam Generator Conditions Observed at Waterford 3	May 20, 2008
WDI-PJF-1304321-FSR-001	Waterford 3 RF16 Reactor Vessel Head Penetration Inspection Final Report	0
DWG C-246-392-2	U.T. Calibration Standard UT-6 (Contract No.74470)	March 14, 1974
CNRO-2007-002	Mitigating Actions and Associated Schedule for Alloy 600/82/182	
Weld No. 12-009	Waterford 3 Dissimilar-Metal Weld Walk-Down Data Sheet 4: 12” SI Nozzle to Safe-End	
	Various Personnel Certifications and Certification Reviews	
	Bare Metal Visual Inspections Scheduled for RF-16	
	RF-16 Steam Generator Scope Summary	

WELDING DATA RECORDS

2009-4293	2009-4528	2009-4588

CONDITION REPORTS

CR-WF3-2006-3966	CR-WF3-2008-2283	CR-HQN-2009-1068	CR-WF3-2009-5194
CR-WF3-2009-5501	CR-WF3-2009-5502	CR-WF3-2009-5509	CR-WF3-2009-5511
CR-WF3-2009-5514	CR-WF3-2009-5515	CR-WF3-2009-5516	CR-WF3-2009-5553
CR-WF3-2009-5554	CR-WF3-2009-5555	CR-WF3-2009-5556	CR-WF3-2009-5585
CR-WF3-2009-5662	CR-WF3-2009-5671	CR-WF3-2009-5679	CR-WF3-2009-5700
CR-WF3-2009-5716	CR-WF3-2009-5735	CR-WF3-2009-5757	CR-WF3-2009-5765
CR-WF3-2009-5769	CR-WF3-2009-5770	CR-WF3-2009-5774	CR-WF3-2009-5836
CR-WF3-2009-5838	CR-WF3-2009-5899	CR-WF3-2009-5941	CR-WF3-2009-5944
CR-WF3-2009-6486	CR-WF3-2009-6504	CR-WF3-2009-6514	CR-WF3-2009-6620

Section 1R11: Licensed Operator Requalification Program

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
EN-TQ-114	Licensed Operator Requalification Training Program Description	0
O-JITDIL	Simulator Scenario for Dilution JIT	3
	Licensed Operator Exam Bank (Parts A and B)	
	Randomly selected licensed operator medical records for five reactor operators and five senior operators	
	All simulator scenarios used for the licensed operator biennial exam	
EN-TQ-200	Training Oversight Program	12
EN-TQ-201	Systematic Approach to Training Process	10
EN-TQ-212	Conduct of Training and Qualification	03
	Productive and Non-Productive [overtime] Report	7/15/2009
	All Licensee Event Reports for 2008 and 2009	7/15/2009
	Waterford 3 Operations Training Comprehensive Assessment Report	5/14/2009
	Operations Training Review Group Meeting Minutes	2008-2009
	All Remedial Training Plans	2008-2009
	Simulator Discrepancy Report	7/15/2009
	Simulator Annual Performance Tests	
OI-024-000	Maintaining Active SRO/RO Status	301
	Weeks 1 and 2 Biennial Written Exams	
	All JPMs Used for the Biennial Exam	
	Various Operator and Operator Training Related Condition Reports	7/13/2009
DG-TRNW-003	Operations Examination Development and Administration	21
DG-TRNW-004	Operations Training Program Lead/Scheduling Desk Guide	31

Focused Assessment – Initial Licensed Operator Training ACAD 02-01, Objectives 2 and 6	9/18/2008
Training Oversight Committee Meeting Minutes	2008/2009

Section 1R12: Maintenance Effectiveness

CONDITION REPORTS

WF3-CR-2008-2637	WF3-CR-2008-2641	WF3-CR-2008-2689	WF3-CR-2008-2721
WF3-CR-2008-3103	WF3-CR-2008-3976	WF3-CR-2008-4012	WF3-CR-2008-4033
WF3-CR-2008-4635	WF3-CR-2008-4953	WF3-CR-2009-2189	WF3-CR-2009-2762
WF3-CR-2009-2796	WF3-CR-2009-3507	WF3-CR-2009-4066	WF3-CR-2009-4088
WF3-CR-2009-4093	WF3-CR-2009-4098	WF3-CR-2009-4155	WF3-CR-2009-5335
WF3-CR-2008-3217	WF3-CR-2008-4992	WF3-CR-2009-1901	WF3-CR-2009-2485
WF3-CR-2008-4453	WF3-CR-2008-5266	WF3-CR-2009-2077	WF3-CR-2009-4499
WF3-CR-2008-4583	WF3-CR-2009-0214	WF3-CR-2009-2096	WF3-CR-2009-5804

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
EN-DC-206	Maintenance Rule	1
NUMARC 93-01	Industry Guideline for Monitoring the Effectiveness of maintenance at Nuclear Power Plants	3

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
EOOS Version 3.3a	Scheduler's Evaluation for Shutdown Version Waterford 3 Rev 3 Model	November 5, 2009
N/A	RF16 Daily Outage Status Report	October 24, 2009
OP-903-107	Surveillance Procedure for Plant Protection System Channel Functional Test	303

EOOS Version 3.3a	Scheduler's Evaluation for Shutdown Version Waterford 3 Rev 3 Model	12/03/2009

Section 1R15: Operability Evaluations

CONDITION REPORTS

CR-WF3-2009-6101 CR-WF3-2008-2684 CR-WF3-2008-2705 CR-WF3-2008-2730

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
EN-OP-104	Operability Determination	4
MI-003-126	Core Protection Calculator Functional	14
SD-PPS	Plant Protection System Description	0
OP-903-107	Plant Protection System Channel A, B, C, D, Functional Test	303
TSTF-324	Correct logarithmic power vs. RTP	1
ECE98-001	Calculation of Maximum Allowable Battery Inter Cell Connection Resistance	0
ECE98-001	Calculation of Maximum Allowable Battery Inter Cell Connection Resistance	1
ME-003-220	Station Battery Bank & Charger (18 month)	303
ME-003-220	Station Battery Bank & Charger (18 month)	301
SD-NI	Nuclear Instrumentation System Description	6

Section 1R19: Postmaintenance Testing

CONDITION REPORTS

CR-WF3-2009-6095	CR-WF3-2009-6412	CR-WF3-2008-2381	CR-WF3-2009-6461
CR-WF3-2009-6449	CR-WF3-2008-4179	CR-WF3-2009-6506	CR-WF3-2009-4499

WORK ORDERS

1517161	213478	187774	152910
161402	122097	212157	

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
STA-001-004	Local Leak Rate Test	303
ICE-37718	Siemens Motor Driven Relay Observed Contact Behavior	02/05/1999
OP-903-116	Train B Integrated Emergency Diesel Generator/Engineering Safety Features Test	013
ME-003-230	Battery Service Test	306
ME-003-240	Battery Performance Test	306
ME-004-213	Battery Intercell Connections	14
ME-004-231	Station Battery Charging	19
ME-003-210	Station Battery Bank and Charger (Quarterly)	16
ME-003-220	Station Battery Bank and Charger (18 month)	303
OP-903-046	Emergency Feed Pump Operability Check - Attachment 10.3	305

Section 1R20: Refueling and Other Outage Activities

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
OP-903-027	Inspection of Containment	301
PLG-009-014	Conduct of Planned Outages	303
OP-001-003	Reactor Coolant System Drain Down	306
OI-037-000	Operations' Risk Assessment Guideline	2
MM-004-201	Containment Building Polar Crane PM	303
WF3-CS-08-01	NEI Heavy Load Drop Initiative	0
UNT-007-008	Control of Loads and Lifting	302
RF-001-009	Reactor Head	303
NEI 08-05	Industry Initiative on Control of Heavy Loads	0
MM-007-003	Containment Building Polar Crane Testing	5

Section 1R22: Surveillance Testing

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
OP-903-116	Train B Integrated Emergency Diesel Generator/Engineering Safety Features Test	013
OP-903-120	Section 7.10 Annulus Negative Pressure Surveillance Test	9

Section 2OS1: Access Controls to Radiologically Significant Areas

CONDITION REPORTS

CR-WF3-2009-5492	CR-WF3-2009-5648	CR-WF3-2009-5878	CR-WF3-2009-5880
CR-WF3-2009-6767	CR-WF3-2009-6792	CR-WF3-2009-6834	CR-WF3-2009-6852
CR-WF3-2009-6856			

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
EN-RP-100	Radworker Expectations	3
EN-RP-101	Access Control for Radiologically Controlled Areas	4
EN-RP-102	Radiological Control	2
EN-RP-105	Radiation Work Permits	6
EN-RP-108	Radiation Protection Posting	7
EN-RP-121	Radioactive Material Control	4
EN-RP-123	Radiological Controls for Highly Radioactive Particles	0
HP-001-114	Control of Temporary Shielding	10
UNT-001-016	Radiation Protection	301
UNT-007-001	Control of Miscellaneous Material in the Spent Fuel Pool	5

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

PROCEDURE/DOCUMENTS

NUMBER	TITLE	DATE
QA-14/15-2009-WF3-1	Radiation Protection/Radwaste Audit	September 2009

RADIATON WORK PERMITS

NUMBER	DESCRIPTION
2009-0401	Perform UDS/Viper/Votes and/or AOV/MOV testing of contaminated system valves
2009-0510	Install/Remove Steam Generator Nozzle Dams, Pin verification, & closeout
2009-0512	Remove/Install Steam Generator Secondary Manways/Handholes
2009-0513	RCP 1A Motor and Driver Mount removal and replacement
2009-0603	Entries into posted LHRA of the Reactor Containment Building to perform minor maintenance activities, walkdowns, surveillances, and inspections
2009-0606	Perform minor maintenance activities, walkdowns, surveillances, and inspections
2009-0628	Entries into Containment Sump to perform transmitter calibrations, Weir Box cleaning and Under Vessel inspections
2009-0721	Entries into posted LHRA of the Reactor Containment Building to install/remove shielding on the ICI stalks
2009-0805	Refuel 16 - Tours and inspections in all RCAs except HRA, LHRA, VHRA

SAMPLE RESULTS AND SURVEYS

MISCELLANEOUS

NUMBER	TITLE	DATE
WF3-0910-0398	Survey of RAB -35 Shutdown Heat Exchangers	October 23, 2009
WF3-0910-0431	Survey of RAB -35 Shutdown Heat Exchangers	October 24, 2009

Section 2OS2: ALARA Planning and Controls

PROCEDURES

NUMBER	TITLE	REVISION
HP-002-201	Radiological Survey Techniques and Frequencies	302
EN-RP-104	Personnel Contamination Events	4
EN-RP-106	Radiological Survey Documentation	2
EN-RP-131	Air Sampling	7
EN-RP-203	Dose Assessment	3

MISCELLANEOUS

NUMBER	TITLE	DATE
2009-0020	Personnel Contamination Event Record	October 29, 2009
2009-0045	Personnel Contamination Event Record	November 3, 2009
2009-0049	Personnel Contamination Event Record	November 5,2009

Section 4OA1: Performance Indicator Verification

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
NEI 99-02	Regulatory Assessment Performance Indicator Guideline	5
EN-LI-114	Performance Indicator Process	4
EN-DIR-RP-002	Radiation Protection Performance Indicator Program	0

MISCELLANEOUS DOCUMENTS

Radiological controlled area entries greater than 100 millirem

Section 4OA2: Identification and Resolution of Problems

CONDITION REPORTS

CR-WF3-2009-5501	CR-WF3-2009-5502	CR-WF3-2009-5509	CR-WF3-2009-5511
CR-WF3-2009-5514	CR-WF3-2009-7166	CR-WF3-2009-7159	

Section 4OA5: Other Activities

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
CEP-NDE-0955	Alloy 600 Visual Examination (VE) of Bare-Metal Surfaces	301
EC-1830	Waterford Steam Electric Station, Unit 3, Dissimilar Metal Weld Overlays	0
Drawing No. WSES-19Q-05	Hot Leg Surge Nozzle Weld Overlay Design	5
SI-UT-130	Procedure for the Phased Array Ultrasonic Examination of Dissimilar Metal Welds	3
SI-NDE-06	Calibration of Ultrasonic NDE Equipment	4
SI-NDE-08	Qualification and Certification of NDE Personnel for Nuclear Applications	1
CEP-NDE-0901	VT-1 Examination	4
CEP-NDE-0902	VT-2 Examination	7
CEP-NDE-0903	VT-3 Examination	5
WF3 11-002 RCP 2A Suction Nozzle	Structural Integrity Associates - Phased Array Ultrasonic Examination Record Data Sheet for Weld No. 11-002: Reactor Coolant Pump 2A Cold Leg Suction Nozzle	October 30, 2009
WF3 11-002 RCP 2A Suction AX SH	Structural Integrity Associates – Ultrasonic Phased Array Calibration Record for Weld No. 11-002: Reactor Coolant Pump Suction Nozzle Dissimilar Metal Weld – Wedge Angle 36.2^{o} (Axial Scan)	October 30, 2009

WF3 11-002 RCP 2A Suction Circ + 10 RL	Structural Integrity Associates – Ultrasonic Phased Array Calibration Record for Weld No. 11-002: Reactor Coolant Pump Suction Nozzle Dissimilar Metal Weld – Wedge Angle 14.0^o (Circumferential Scan)	October 30, 2009
WF3 11-002 RCP 2A Suction Flat RL	Structural Integrity Associates – Ultrasonic Phased Array Calibration Record for Weld No. 11-002: Reactor Coolant Pump Suction Nozzle Dissimilar Metal Weld – Wedge Angle 14.0^o (Axial & Circumferential Scan)	October 30, 2009
WF3-LIN-09-002	Structural Integrity Associates – Ultrasonic Linearity Record – Zetec/RD Tech OmniScan MX – Version 1.4R3 (Serial No. ONMI-1983); Transducer 115-000-613 (Serial No. 01VTVW); Reference Block 16" AX (Serial No. SI-16-AX-03).	October 21, 2009
Contract No. C-09-089 R1	Sonaspection – Structural Integrity of Calibration Block No. SI-Flat-SS-4inchT-01	May 18, 2009