UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BOULEVARD, SUITE 100
KING OF PRUSSIA, PENNSYLVANIA 19406-2713

February 11, 2013

Mr. Kevin Walsh
Site Vice President
Seabrook Nuclear Power Plant
NextEra Energy Seabrook, LLC
c/o Mr. Michael O'Keefe
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 - NRC INTEGRATED INSPECTION REPORT 05000443/2012005

Dear Mr. Walsh:

On December 31, 2012, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at Seabrook Station, Unit No. 1. The enclosed inspection report documents the inspection results, which were discussed on January 2, 2013, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two self revealing findings of very low safety significance (Green). 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 (CAP), the NRC is treating these findings as non-cited violations (NCVs), consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest any NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Seabrook Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspector at Seabrook Station.

In accordance with 10 Code of Federal Regulations (CFR) 2.390 of the NRCs "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records component of the NRC's Agencywide Documents Access Management System (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.htmL (the Public Electronic Reading Room).

Sincerely,

/RA/

Daniel L. Schroeder, Acting Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket No. 50-443
License No: NPF-86

Enclosure: Inspection Report No. 05000443/2012005
w/ Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-443
License No.:	NPF-86
Report No.:	05000443/2012005
Licensee:	NextEra Energy Seabrook, LLC
Facility:	Seabrook Station, Unit No.1
Location:	Seabrook, New Hampshire 03874
Dates:	October 1, 2012, through December 31, 2012
Inspectors:	S. Rich, Acting Senior Resident Inspector M. Jennerich, Resident Inspector T. O'Hara, Reactor Inspector B. Dionne, Health Physics Inspector D. Silk, Senior Operations Engineer C. Newport, Reactor Inspector
Approved by:	Daniel L. Schroeder, Acting Chief Reactor Projects Branch 3 Division of Reactor Projects

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SUPPLEMENTARY INFORMATION A-1

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SUMMARY OF FINDINGS

IR 05000443/2012005; 10/1/2012-12/31/2012; Seabrook Station, Unit No. 1; Post Maintenance Testing.

This report covered a three-month period of inspection by resident inspectors and announced inspections performed by regional inspectors. Two self-revealing findings and one licensee-identified finding of very low safety significance (Green), were identified and dispositioned as non-cited violations (NCVs). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross-cutting aspects for the findings were determined using IMC 0310, "Components Within Cross-Cutting Areas." Findings for which the SDP does not apply may be Green, or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

Cornerstone: Initiating Events

- Green: A self revealing, non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," was identified because the high pressure Swagelok fitting for the L-5 fixed in-core detection instrument failed and caused an unisolable reactor coolant leak. Specifically, NextEra did not implement timely and effective corrective actions to address a degraded Swagelok fitting associated with the L5 in-core instrument connection that was identified as a condition adverse to quality in 2006. The inspectors determined that not taking timely and effective corrective action to correct a condition adverse to quality was a performance deficiency. The inadequate corrective actions led to the failure of the L-5 fixed in-core detection instrument Swagelok fitting on October 21, 2012. The inspectors determined that this issue was within NextEra's ability to foresee and correct, because this fitting was identified as leaking during previous operating cycles, was assigned additional monitoring and the adverse trend of increased leakage at L-5 at low pressures continued from the time it was identified in 2006. NextEra entered this into their corrective action program as AR 01815351 and implemented immediate corrective actions to cut the connection for the L-5 instrument, as well as two others showing signs of leakage, and capped the tubes prior to recommencing start-up.

  The inspectors determined that the performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone's objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Additionally it is similar to example 4.d of Inspection Manual (IMC) 0612, Appendix E, because this was a failure to implement a corrective action that did have a safety impact, because the fitting failed and caused a 4 gpm non-isolable leak from the reactor coolant system. The inspectors evaluated the finding using IMC 0609, Attachment A, because the operational impact occurred after the residual heat removal pump was secured for start-up. The inspectors determined that the finding was of very low safety significance (Green) because the deficiency would not result in exceeding the small loss of coolant accident (LOCA) leak rate and would not have affected other systems used to mitigate a LOCA.

  This finding has a cross-cutting aspect in the area of Human Performance, Resources, because actions were not taken to maintain long term plant safety by minimization of long-standing equipment issues. Specifically, NextEra did not manage the ongoing degradation of the L-5 in-core instrument connection fitting connection while long term corrective actions were implemented. [H.2(a)] (Section 1R19)

- Green: A self-revealing, non-cited violation of technical specification 6.7.1, "Procedures and Programs," was identified after the control room received a high discharge temperature alarm for pressurizer relief valve RC-V-116 while pressurizing the reactor coolant system during start-up preparations on October 21, 2012. Specifically, NextEra personnel did not properly implement maintenance procedure MS0519.17, "Crosby Pressurizer Mechanical Safety Valve Removal and Installation." This led to the reactor coolant system leakage past the RC-V-116 flange gasket that caused the high discharge temperature alarm. The inspectors determined that not properly implementing procedure MS0519.17 was a performance deficiency that was within NextEra's ability to foresee and correct. NextEra entered this into their corrective action program as AR1815307 and implemented immediate corrective actions to retorque the bolts and replace the gasket on RC-V-116.

  The performance deficiency was determined to be more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, because NextEra personnel did not properly implement procedure MS0519.17, eight bolts on the inlet flange of pressurizer RC-V-116 were not adequately torqued. This resulted in reactor coolant system leakage during preparations for reactor start-up on October 21, 2012, and required NextEra operators to return the plant to cold shutdown. Additionally, this was similar to more-than-minor example 2.e in IMC 0612, Appendix E, because the procedure non-compliance resulted in a negative safety consequence in that it impacted the ability of the flange to perform its function to prevent reactor coolant system leakage. The inspectors evaluated the finding using IMC 0609, Attachment A, because the operational impact occurred after the residual heat removal pump was secured for start-up. The inspectors determined that the finding was of very low safety significance (Green) because the deficiency would not result in exceeding the small loss of coolant accident (LOCA) leak rate and would not have affected other systems used to mitigate a LOCA.

  This finding has a cross-cutting aspect in the area of Human Performance, work practices, because personnel did not follow the procedures. Specifically, when tensioning the bolts on the pressurizer relief valve RC-V-116 inlet flange, NextEra personnel did not verify there was a gap for eight of the twelve bolts on the inlet flange of the valve as required by maintenance procedure MS0519.17. [H.4(b)] (Section 1R19)

Other Findings

- One violation of very low safety significance that was identified by NextEra was reviewed by the inspectors. Corrective actions taken and planned by NextEra have been entered into NextEra's corrective action program. The violation and corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Seabrook started the period in a planned refueling outage. The reactor achieved criticality on October 29, 2012, and reached 100% power on November 2, 2012. On December 31, 2012, Seabrook performed a downpower to 50 percent to perform maintenance on the B main feed pump. Later that day, power was increased to 91 percent as part of the return to full power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 – 2 samples)

.1 Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

On October 28, 2012, the inspectors reviewed NextEra's preparations for potential tropical storm wind conditions and heavy rain associated with Hurricane Sandy. The inspectors reviewed the entry criteria for and actions specified in OS1200.03, "Severe Weather Conditions," Revision 19. The inspectors performed a walkdown of the protected area to verify items were tied down or stored so they would not be affected by the high winds, and performed a walkdown of inside areas of the plant to verify there was no excessive water intrusion. Documents reviewed for each section of this inspection report are listed in the Attachment.

b. Findings

No findings were identified.

.2 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

During the week of December 17, 2012, the inspectors performed a review of NextEra's readiness for seasonal cold temperatures. The review focused on the Service Water Cooling Tower and Fire Pump House heating systems. The inspectors reviewed the Updated Final Safety Analysis report (UFSAR), technical specifications, applicable procedures and design documents to determine what temperatures or other seasonal weather could challenge these systems, and to ensure that NextEra personnel had implemented the station procedures for seasonal readiness. The inspectors performed walk downs of the selected systems to ensure station personnel identified issues that could challenge operability of the systems during cold weather.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns (71111.04Q – 2 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- A emergency feed water (EFW) system on November 13, 2012
- B emergency diesel generator (EDG) system with switchyard work ongoing on December 21, 2012

The inspectors selected these systems based on their risk-significance in the current plant configuration. The inspectors reviewed applicable operating procedures, system diagrams, the UFSAR, technical specifications, work orders, condition reports, and the impact of ongoing work activities in order to identify conditions that could have impacted system performance of their intended safety functions. The inspectors also performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and were operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies.

b. Findings

No findings were identified.

.2 Full System Walkdown (71111.04S – 1 sample)

a. Inspection Scope

On October 31, 2012, and November 2, 2012, the inspectors performed a complete system walkdown of accessible portions of the service water system to verify the equipment lineup was correct. The inspectors reviewed operating procedures, surveillance tests, drawings, equipment lineup procedures, and the UFSAR to verify the system was aligned to perform its required safety functions. The inspectors also reviewed electrical power availability, component lubrication and equipment cooling, hangar and support functionality, and operability of support systems. The inspectors performed field walkdowns of accessible portions of the system 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 the equipment to verify that there were no deficiencies. The inspectors also reviewed whether NextEra staff had properly identified equipment issues and entered them into the corrective action program for resolution with the appropriate significance characterization. Additionally, the inspectors reviewed a sample of related condition reports and work orders to ensure NextEra appropriately evaluated and resolved any deficiencies.

b. Findings

No findings were identified.

1R05 Fire Protection

.1 Resident Inspector Quarterly Walkdowns (71111.05Q – 4 samples)

a. Inspection Scope

The inspectors conducted tours of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that NextEra controlled combustible materials and ignition sources in accordance with administrative procedures. The inspectors verified that fire protection and suppression equipment was available for use as specified in the area pre-fire plan, and passive fire barriers were maintained in good material condition.

- EDG building oil tank room train B on November 13, 2012
- EFW pump house on November 13, 2012
- A residual heat removal (RHR) equipment vault on November 21, 2012
- B RHR equipment vault on December 27, 2012

1R06 Flood Protection Measures (71111.06 – 1 sample)

.1 Annual Review of Cables Located in Underground Bunkers/Manholes

a. Inspection Scope

The inspectors conducted an inspection of underground manholes subject to flooding that contain safety-related cables. The inspectors observed NextEra staff checking manhole water level on two occasions, dewatering as necessary, and verified through NextEra's cable monitoring program documents and past photographs of all six manholes subject to flooding that the identified water level was below the level of the cables. The inspectors also reviewed two past work orders to verify that NextEra pumped water out of the manholes before the level became deep enough to submerge the cables.

b. Findings

No findings were identified.

1R08 In-service Inspection (71111.08 – 1 sample)

a. Inspection Scope

From September 17 through September 21, and between September 30 and October 5, the inspectors conducted a review of NextEra's implementation of in-service inspection (ISI) program activities for monitoring degradation of the reactor coolant system pressure boundary, risk significant piping and components, and containment systems for Seabrook Station Unit 1. The sample selection was based on the inspection procedure objectives and risk priority of those pressure retaining components in these systems where degradation would result in a significant increase in risk. The inspectors observed in-process non-destructive examinations (NDE), reviewed documentation, and interviewed inspection personnel to verify that the non-destructive examination activities performed as part of Period 1 of the Third 10 year Interval of the Seabrook Unit 1 In-service Inspection Program during outage OR15 were conducted in accordance with the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, 2004 Edition, No Addenda.

Nondestructive Examination and Welding Activities (Section 02.01)

The inspectors performed direct observations of NDE activities in process and reviewed records of nondestructive examinations listed below:

ASME Code Required Examinations

- Reviewed the ultrasonic (UT) records of the inspection of weld overlay RC E-10 S-SWOL-ISI and the UT thickness examinations of degraded service water piping, and observed the inspections and reviewed the pre-service UT records of safety injection system welds 0251-07-04, 0251-07-05, and 0251-07-09.

- Documentation review of the bare metal visual examination of the reactor vessel upper closure head and control rod drive mechanism nozzle penetrations.

- Reviewed the visual testing (VT) procedures and the recorded inspection results for the upper dome portion of the containment liner. These inspections were conducted in accordance with the ASME Code Section XI, Subsection IWE.

- Reviewed the magnetic particle testing (MT) data sheets from several inspections of degraded service water piping.

- Reviewed the visual inspection records for the visual inspection of bottom mounted instrumentation tubes in the reactor vessel bottom head.

- Remote observation of steam generator (SG) eddy current testing (ECT) examinations, data evaluation, and documentation review of the final list of pluggable tubes from this inspection.

- Reviewed the records of the independent general visual ASME, IWE inspection of Seabrook Unit 1 containment liner coating during OR15.

The inspectors reviewed certifications of the NDE technicians performing the examinations and verified that the inspections were performed in accordance with approved procedures and that the results were reviewed and evaluated by certified Level III NDE personnel.

Other Augmented IWE Examinations

The inspectors reviewed UT examination results completed by NextEra to determine whether a condition (alkali-silica reaction) observed on the external surface of the reinforced concrete containment support structure resulted in any deterioration of the containment carbon steel liner (pressure boundary). NextEra performed the UT thickness examination on a one-foot by one-foot area of the accessible interior surface of the containment liner at elevation minus 26-foot and approximate azimuth of 317.5

degrees. The containment liner UT results identified this area to be of nominal wall thickness (0.375 inches), with no exterior or interior surface degradation detected. The UT examination results were documented in work order number 40185220-01.

Review of Originally Rejectable Indications Accepted by Evaluation

- The inspectors reviewed a volumetric examination data sheet for a rejectable indication on a previous inspection of weld RC RPV SE-301-121-H. The engineering evaluation accepted the indication for further service based upon the requirements of the ASME Code.

- The inspectors reviewed the engineering evaluation of an originally rejectable indication on a previous inspection of repair RC E-10 X-SWOL, a structural weld overlay on a pressurizer surge line weld. The evaluation accepted the indication for further service based upon the requirements of the ASME Code.

- The inspectors reviewed the evaluation of an originally rejectable indication from a prior inspection of feedwater pipe weld FW 4608-11 06. The engineering evaluation accepted the indication for further service based upon the requirements of the ASME Code.

Repair/Replacement Consisting of Welding Activities

The inspectors reviewed replacement activities for the replacement of safety injection system, loop 3, injection check valve 1-SI-V-82. The corrective action process documented a history of body-to-bonnet gasket leaks from this valve. The inspectors reviewed Engineering Change Number 0000272512-002, 10/13/2011, which engineered the replacement of the valve during OR15. The inspectors reviewed the radiographic inspection report for the weld inspection of shop welds and piping pieces in preparation for field welding the valve into the safety injection system piping. The inspectors also observed the welding of the replacement valve assembly and the preservice UT inspections of the new welds for the replacement valve. The inspectors reviewed the flow surveillance test performed to verify the functioning of the check valve. The inspectors also verified that the new valve was tested at system operational pressure before plant startup. This repair was completed in accordance with ASME, Section XI, Repair/Replacement Requirements.

Additionally, the inspectors reviewed the ASME, Section XI repair of a leak in the service water system common header, an ASME, Class III component. The NextEra engineering change required and internal patch be welded over a leaking, degraded section of the common header. Welding of the patch to the piping, nondestructive examination of the weld and post repair testing of the piping was completed satisfactorily in accordance with the ASME Code, Section XI, Repair/Replacement Requirements.

During these repair and replacement activities the inspectors verified a sample of the welder qualifications for the welders performing the repair.

Reactor Vessel Upper Head Penetration Inspection Activities (Section 02.02)

The inspectors reviewed calculations of effective degradation years and re-inspection year for the reactor vessel upper head. Based on these calculation results, NextEra deferred the inspection of the reactor vessel upper head penetration J-groove weld examinations until the next scheduled refueling outage, OR16. These calculations were performed in accordance with the requirements of 10 CFR 50.55a(g)(6)(ii)(D) and the ASME Boiler and Pressure Vessel Code Case N-729-1, “Alternative Examination Requirements for PWR Reactor Vessel Upper Heads,” to ensure the structural integrity of the reactor vessel head pressure boundary. The inspectors reviewed visual inspection reports of the remote bare metal visual examination of the exterior surface of Unit 1 reactor vessel upper head penetrations to confirm appropriate inspection coverage was achieved and to verify that no boric acid leakage or wastage was observed.

Boric Acid Corrosion Control Inspection Activities (Section 02.03)

The inspectors reviewed the boric acid corrosion control program, which is performed in accordance with site and corporate procedures, discussed the program with the boric acid program owner, and sampled photographic inspection records of boric acid found on risk significant piping and components inside the containment during walkdowns conducted by NextEra personnel and observed by the NRC Resident Inspectors on September 24, 2012. The inspectors observed the identification and documentation of non-conforming conditions of boric acid leaks in the corrective action program with emphasis on areas that could cause degradation of safety significant components.

The inspectors verified that potential deficiencies identified during the walkdowns were entered into NextEra’s corrective action program and reviewed a sample of engineering evaluations for the conditions to verify that the corrective actions were consistent with the requirements of the NextEra procedures and 10 CFR 50, Appendix B, Criterion XVI. The inspectors also reviewed the associated engineering evaluations for the above condition reports to verify that equipment or components that were wetted or impinged upon by boric acid solutions were properly analyzed for degradation.

Steam Generator Tube Inspection Activities (Section 02.04)

The inspectors directly observed a sample of the steam generator (SG) eddy current tube examinations, which consisted of full length bobbin inspection of 100 percent of the in-service tubes in each of the four SGs, +Pt inspection of 50 percent of Row 1 and 2 U-bends, +Pt inspection of 50 percent of dings/dents >5V on the hot leg including the U-bend, +Pt inspection of 50 percent hot leg, top of the tube sheet plus 3-inches (to include 50 percent of the OXP/BLG within the H* depth), visual inspection of all plugs (mechanical and welded), and channel head visual inspection per NSAL-12-1. The inspectors reviewed a sample of the indications identified in the SGs during the prior outage ECT inspections to verify that they were consistent with the potential degradation mechanisms that may be observed during the OR15 inspections as documented in SG-SGMP-12-8, Revision 1, “Steam Generator Degradation Assessment for Seabrook OR15 Refueling Outage.” The inspectors also reviewed the Condition Monitoring and Operational Assessments from the prior outage.

The inspectors verified that the SG eddy current tube examinations were performed in accordance with Unit 1 Technical Specifications, Unit 1 Steam Generator Program by reviewing the SG tube ECT results to verify that no in-situ pressure testing was required, no tubes required stabilization, no primary-to-secondary leakage occurred over the operating cycle 14, and the tubes that exhibited degradation and did not meet acceptance criteria were plugged (9 tubes) or sleeved (0 tubes) using the alternate repair criteria per Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," during the OR15 inspection. The inspectors verified that the SG tube examination screening criteria was in accordance with the Electric Power Research Institute (EPRI) Steam Generator Guidelines and flaw sizing was in accordance with EPRI examination technique specification sheet.

During OR15, NextEra plugged an additional three tubes in SG A for anti-vibration bar (AVB) wear greater than 40 percent; an additional tube in SG B for a single axial indication (SAI) and one additional tube for a stuck probe head; an additional two tubes in SG C for AVB wear greater than 40 percent and one for an SAI, and one additional tube plugged in SG D for AVB wear greater than 40 percent. A total of 182 tubes were plugged in all four SG's at the completion of OR15.

When the SAI was identified in SG A and the SAI in SG B, NextEra expanded their inspection program as required by the EPRI SG Guidelines. The expanded scope did not identify additional SAI indications.

Also, the inspectors reviewed the foreign object search and retrieval results on the secondary side of the SGs, and reviewed corrective actions to remove the foreign objects if possible.

The inspectors reviewed the Westinghouse eddy current testing procedures and verified that NextEra completed the steam generator inspections in accordance with the requirements of Nuclear Energy Institute 97-06, "Pressurized Water Reactor Steam Generator Examination Guidelines"; Revision 7. The inspectors also reviewed the Westinghouse procedure qualification certifications and a sample of data analysts' personnel certifications. The inspectors reviewed a sample of eddy current qualification records for Primary and Secondary Resolution Analysts, Independent Quality Data Analysts, and Utility Level III Quality Data Analysts.

Identification and Resolution of Problems (Section 02.05)

The inspectors reviewed a sample of condition reports, which identified NDE indications, deficiencies and other nonconforming conditions since the previous outage and during the OR15 outage. The inspectors verified that nonconforming conditions were properly identified, characterized, evaluated, corrective actions identified and dispositioned, and appropriately entered into the corrective action program.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11Q – 2 samples)

.1 Quarterly Review of Licensed Operator Requalification Testing and Training

a. Inspection Scope

The inspectors observed licensed operator simulator training on November 19, 2012, which included a power range nuclear instrument failure, with an automatic rod insertion, leading to a reactor trip. The inspectors evaluated operator performance during the simulated event and verified completion of risk significant operator actions, including the use of abnormal and emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, implementation of actions in response to alarms and degrading plant conditions, and the oversight and direction provided by the control room supervisor. The inspectors verified the accuracy and timeliness of the emergency classification made by the shift manager and the technical specification action statements entered by licensed operators. Additionally, the inspectors assessed the ability of the crew and training staff to identify and document crew performance problems.

b. Findings

No findings were identified.

.2 Quarterly Review of Licensed Operator Performance in the Main Control Room

a. Inspection Scope

The inspectors observed and reviewed start-up surveillance activities, including low power physics testing and turbine overspeed testing conducted on October 29-30, 2012. The inspectors observed pre-job briefings to verify that the briefings met the criteria specified in OP-AA-100-1000, “Conduct of Operations,” Revision 8. Additionally, the inspectors observed test performance to verify that procedure use, crew communications, and coordination of activities between operators and reactor engineers met established expectations and standards.

a. Findings

No findings were identified.

.3 Licensed Operator Requalification (71111.11A – 1 sample)

a. Inspection Scope

On December 27, 2012, a region-based inspector conducted an in-office review of results of the licensee-administered comprehensive written exams and annual operating tests. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process (SDP).” The inspector verified that:

- Individual pass rate on the dynamic simulator test was greater than 80 percent (The pass rate was 100 percent)

- Individual pass rate on the job performance measures of the operating exam was greater than 80 percent (The pass rate was 100 percent)
- Individual pass rate on the written examination was greater than 80 percent (There was no written exam in 2012. The written exam pass rate in 2011 was 98.1 percent)
- More than 80 percent of the individuals passed all portions of the exam (The pass rate was 100 percent)
- Crew pass rate was greater than 80 percent (The pass rate was 100 percent)

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12 – 2 samples)

a. Inspection Scope

The inspectors reviewed the samples listed below to assess the effectiveness of maintenance activities on systems, structures and components (SSCs) performance and reliability. The inspectors reviewed system health reports, maintenance backlogs, and maintenance rule (MR) basis documents to ensure that NextEra was identifying and properly evaluating performance problems within the scope of the MR. For each sample selected, the inspectors verified that the SSC was properly scoped into the maintenance rule in accordance with 10 CFR 50.65 and verified that the (a)(2) performance criteria established by NextEra staff was reasonable. Additionally, the inspectors ensured that NextEra staff was identifying and addressing common cause failures that occurred within and across maintenance rule system boundaries.

- Service Water Cooling Tower
- Nuclear Instruments

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 4 samples)

a. Inspection Scope

The inspectors reviewed station evaluation and management of plant risk for the planned maintenance and emergent work activities listed below to verify that NextEra personnel performed risk assessments as required by 10 CFR 60.65(a)(4) and applicable station procedures, and that the assessments were accurate and complete. The inspectors selected these activities based on potential risk significance. The inspectors verified that NextEra performed the appropriate risk assessments prior to removing equipment for work. When NextEra performed emergent work, the inspectors verified that operations personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work and discussed the results of the assessment with the station's probabilistic risk analyst to verify 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.

- Planned maintenance to install S/G Dams for outage work, requiring mid-loop operations on October 11, 2012
- Emergent maintenance on the service water (SW) tee common discharge header with A SW secured on October 18, 2012
- Planned maintenance for service water piping repairs on October 24, 2012
- Emergent maintenance due to SW Train B unavailable due to screen wash issues, November 26, 2012, to December 2, 2012

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 – 4 samples)

a. Inspection Scope

The inspectors reviewed operability determinations for the following degraded or non-conforming conditions:

- CR 01808748, High Vibrations on A EDG Crankcase Exhaust Fan (DG-FN-29-A)
- CR 01813095, DG-1B Maximum Voltage Criteria Not Met in B engineered safety features (ESF) testing
- CR 01813412, Service Water Pipe Liner Delamination
- CR 01815307, Reactor Coolant System (RCS) Leak due to RC-V-116 Loose Flange Bolting

The inspectors selected these issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the operability determinations to assess whether 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 UFSAR to NextEra'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 by NextEra. The inspectors determined, where appropriate, compliance with assumptions in the evaluations.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18 – 2 samples)

.1 Permanent Modifications

a. Inspection Scope

The inspectors evaluated a modification to the A EDG engine temperature control system for the jacket water and air cooler implemented by engineering change EC-144992, "EDG Engine Temperature Control Upgrade". The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modification. The inspectors also reviewed revisions to the control room alarm response procedure and interviewed engineering and operations personnel to ensure the procedure could be reasonably performed. The inspectors observed the operation of the A EDG during testing of the upgraded temperature control system, and also the technical specification required surveillance to ensure operability of the system.

The inspectors evaluated a modification to Service Water Piping Liner repairs implemented by engineering change EC-274750, "Liner Repair Options for Cement Lining in Service Water Piping. The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modification. The inspectors verified that the lining repair method was adequate and capable of performing the design function required of the system.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 – 6 samples)

a. Inspection Scope

The inspectors reviewed the post-maintenance tests for the maintenance activities listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the test procedure to verify that the procedure adequately tested the safety functions that may have been affected by the maintenance activity, that the acceptance criteria in the procedure was consistent with the information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed test data to verify that the test results adequately demonstrated restoration of the affected safety functions.

- Emergency feedwater valve 1-FW-FV-4214-B multiple spurious operations modification testing on October 18, 2012
- Service water tee leak repair on October 19, 2012
- Pressurizer safety valve RC-V-116 replacement on November 2, 2012
- Seal table repairs on November 2, 2012
- Replacement of 1-SW-PT-8283 on November 28, 2012
- Charging pump A breaker inspection on December 13, 2012

b. Findings

.1 Introduction: A self revealing, Green, non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," was identified because the high pressure Swagelok fitting for the L-5 fixed in-core detection instrument failed and caused an unisolable reactor coolant leak. Specifically, NextEra did not implement timely and effective corrective actions to address a degraded Swagelok fitting associated with the L5 in-core instrument connection that was identified as a condition adverse to quality in 2006. As a result, the fitting continued to degrade and failed on October 21, 2012.

Description: On October 21, NextEra was in the process of heating up and pressurizing the reactor coolant system following refueling outage 15 (OR15). Operators received the "Reactor Sump A or B In-Leakage High" alarm. NextEra commenced returning the plant to a cold shutdown condition. The operators calculated an approximate reactor coolant system (RCS) leak rate of 6 gallons per minute based on the rate of change of the reactor sump level. Thirty minutes after the alarm was received in the control room, field operators reported steam coming from the in-core instrumentation seal table.

Following the plant shutdown, it was determined that the high pressure Swagelok fitting for the L-5 connection of the fixed in-core detection instrument seal table had failed and partially ejected the instrument approximately 2 ft. This caused an unisolable reactor coolant leak from the reactor vessel. NextEra personnel determined that the actual RCS leak was 4 gpm.

The inspectors reviewed maintenance history and identified several instances of leaking in-core detection instrument fittings at Seabrook. Since 2006; NextEra had classified the active leak at the L-5 connection at the seal table as a condition adverse to quality. This fitting was not replaced at that time because replacement of a leaking Swagelok fitting was not a viable solution until the detector needed replacement because the process to replace Swagelok fittings required the detector to be removed and a new detector installed. As a result of satisfactory performance of the in-core detectors, it was not until 2007 that NextEra initiated plans to replace all of the in-core detectors based on detector age and performance. The priority assignments for detector replacement in this plan did not consider the impact of or condition of the Swagelok fittings that were leaking.

Following OR14 in 2011, NextEra identified increased leakage at the seal table on multiple occasions, including at the L-5 connection. In response, NextEra created a preventative maintenance task to monitor the seal table monthly. This activity included taking pictures and documenting the connections that were leaking; however, no limits were established to direct additional action to be taken, no interim actions were taken to ensure that identified leakage did not get worse, and no actions were taken to expedite detector and fitting replacement based on the increased leakage rate at a fitting.

The trend in leakage from the L-5 instrument connection continued to degrade during the operating cycle leading up to OR15. In OR15, NextEra replaced several detectors, but not the L-5 detector, because they prioritized replacement based on detector function only and did not consider the degraded connections when choosing which to replace. NextEra took no action to correct the leaking L-5 fitting. As a result, while pressurizing prior to commencing the start-up after OR15, NextEra personnel identified that the L-5 fitting was leaking under an RCS pressure of 46 psig, 250 psig, and 850 psig, and the fitting failed and the detector was ejected at approximately 1400 psig.

The inspectors conducted a review of industry operating experience regarding this issue and determined that historically these connections were known to leak and that there were multiple instances of industry operating experience including NRC Information Notice No. 84-55 that described the potential for detector ejection as a result of Swagelok fitting leakage. A review of corrective action history did not identify any actions by NextEra to address this potential related to the leakage that they were experiencing in the in-core instrumentation fittings.

NextEra entered this issue into their corrective action program as AR 01815351 and implemented immediate corrective actions to cut the connection for the L-5 instrument, as well as two others showing signs of leakage, and restored the reactor coolant pressure boundary by capping the affected tubes. The inspectors determined that delaying long term corrective actions and the lack of effective interim corrective actions to manage the degrading condition adverse to quality associated with the L-5 in-core instrument connection fitting resulted in the failure of the reactor pressure boundary at the fitting on October 21.

Analysis: The inspectors determined that not taking timely and effective corrective action to correct a condition adverse to quality was a performance deficiency. The inadequate corrective actions led to the failure of the L-5 fixed in-core detection instrument Swagelok fitting on October 21. The inspectors determined that this issue was within NextEra's ability to foresee and correct, because this fitting was identified as leaking during previous operating cycles, was assigned additional monitoring and the adverse trend of increased leakage at L-5 at low pressures continued from the time it was identified in 2006. The inspectors determined that the performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone's objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Additionally it is similar to example 4.d of Inspection Manual (IMC) 0612, Appendix E, because this was a failure to implement a corrective action that did have a safety impact, because the fitting failed and caused a 4 gpm non-isolable leak from the reactor coolant system. The inspectors evaluated the finding using IMC 0609, Attachment A, because the operational impact occurred after the residual heat removal pump was secured for start-up. The inspectors determined that the finding was of very low safety significance (Green) because the deficiency would not result in exceeding the small loss of coolant accident (LOCA) leak rate and would not have affected other systems used to mitigate a LOCA.

This finding has a cross-cutting aspect in the area of Human Performance, Resources, because actions were not taken to maintain long term plant safety by minimization of long-standing equipment issues. Specifically, NextEra did not manage the ongoing degradation of the L-5 in-core instrument connection fitting connection while long term corrective actions were implemented. [H.2(a)]

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, requires, in part, that conditions adverse to quality are promptly identified and corrected. Contrary to this requirement, NextEra did not correct an identified condition adverse to quality. Specifically, between 2006 and October 21, 2012, NextEra did not take timely and effective corrective actions to correct an identified reactor coolant leak that it had identified at the L-5 in-core instrument Swagelok fitting. As a result, the fitting continued to degrade and failed on

October 21, 2012, requiring operators to place the plant in cold shutdown. Because this issue is of very low safety significance (Green) and NextEra entered this into their corrective action program as AR 01815351 and implemented immediate corrective actions to cut the connection for the L-5 instrument, as well as two others showing signs of leakage, and capped the tubes prior to recommencing start-up, this finding is being treated as an NCV consistent with the NRC Enforcement Policy. **(NCV 05000443/2012005-01, Failure to Correct a Condition Adverse to Quality for the L-5 FICI Connection)**

.2 Introduction: A self-revealing, Green, NCV of technical specification 6.7.1, "Procedures and Programs," was identified after the control room received a high discharge temperature alarm for pressurizer relief valve RC-V-116 while pressurizing the reactor coolant system during start-up preparations on October 21. Specifically, NextEra personnel did not properly implement maintenance procedure MS0519.17, "Crosby Pressurizer Mechanical Safety Valve Removal and Installation." This led to the reactor coolant system leakage past the RC-V-116 flange gasket that caused the high discharge temperature alarm.

Description: On October 21, while pressurizing the reactor coolant system in preparation for start-up, the control room received an alarm for high discharge temperature for RC-V-116 at about 1400 psig. NextEra entered their abnormal operating procedure and returned the plant to cold shutdown. Upon inspection, NextEra personnel noted trace amounts of boric acid on the flanges of all three pressurizer safety valves, indicating that there was reactor coolant leakage. NextEra personnel checked the torque on the flange nuts of the affected valves and discovered that out of the twelve total nuts on RC-V-116, four were left hand-tight and four were below the required torque specification. The other two pressurizer relief valves, RC-V-115 and RC-V-117, also had bolts that were not fully torqued, but the effects on these valves were more limited than those on RC-V-116. NextEra performed an evaluation and determined that leakage through the flange of RC-V-116 would likely have exceeded the 1 gallon per minute limit for unidentified leakage had the valve been at full pressure, but the joint would not have experienced gross failure.

NextEra's review of the maintenance history for these valves determined that NextEra personnel had installed pressurizer safety valve RC-V-116 on September 26. The bolts on the inlet flange of the pressurizer safety valves are tensioned using a system that stretches all the bolts simultaneously using hydraulic pistons. The hydraulic pressure source for this system connects to each of the bolt pistons individually, and, therefore, each line contains a valve that prevents the flow of fluid to apply tension to the bolts unless it is properly connected to the piston. The best indication that a line is properly connected and that the bolt is being tensioned is the formation of a gap between the hydraulic piston and the lock ring that is installed on the bolt to maintain tension after the system's hydraulic pressure is removed. Procedure MS0519.17, "Crosby Pressurizer Mechanical Safety Valve Removal and Installation," includes a step that requires the user of the tension system to verify that this gap is formed. NextEra's review of the maintenance and restoration for RC-V-116 determined that the NextEra staff performing the installation did not properly verify there was a gap for some of the bolts on the inlet flange of the valve and therefore did not identify that there were improper connections for four of the hydraulic lines for the tension system. The improper connections resulted in eight bolts not being properly torqued during reinstallation.

As immediate corrective action for the identified conditions, NextEra retorqued the loose bolts identified on RC-V-115 and RC-V-117 and replaced the gasket and retorqued the bolts on RC-V-116. The three valve flanges were then checked for leakage at full system pressure and verified to be tight. NextEra also performed a root cause evaluation for the event and determined that, while the procedure had weaknesses, the root cause of the performance deficiency was failure to follow procedures. This conclusion was based on the fact that the same procedure for removal and reinstallation of pressurizer mechanical safety valves was used successfully during four previous outages.

Analysis: The inspectors determined that not properly implementing procedure MS0519.17 was a performance deficiency that was within NextEra's ability to foresee and correct. The performance deficiency was determined to be more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, because NextEra personnel did not properly implement procedure MS0519.17, eight bolts on the inlet flange of pressurizer RC-V-116 were not adequately torqued. This resulted in reactor coolant system leakage during preparations for reactor start-up on October 21, and required NextEra operators to return the plant to cold shutdown. Additionally, this was similar to more-than-minor example 2.e in IMC 0612, Appendix E, because the procedure non-compliance resulted in a negative safety consequence in that it impacted the ability of the flange to perform its function to prevent reactor coolant system leakage. The inspectors evaluated the finding using IMC 0609, Attachment A, because the operational impact occurred after the residual heat removal pump was secured for start-up. The inspectors determined that the finding was of very low safety significance (Green) because the deficiency would not result in exceeding the small loss of coolant accident (LOCA) leak rate and would not have affected other systems used to mitigate a LOCA.

This finding has a cross-cutting aspect in the area of Human Performance, work practices, because personnel did not follow the procedures. Specifically, when tensioning the bolts on the pressurizer relief valve RC-V-116 inlet flange, NextEra personnel did not verify there was a gap for eight of the twelve bolts on the inlet flange of the valve as required by maintenance procedure MS0519.17. [H.4(b)]

Enforcement: Technical Specification 6.7.1, "Procedures and Programs," requires, in part, that procedures recommended in Appendix A of Regulatory Guide 1.33, Rev. 2, be implemented. RG 1.33, Appendix A, Section 9, includes procedures for performing maintenance. Contrary to the above, on September 26, NextEra did not adequately implement procedures for performing maintenance on pressurizer safety valves. Specifically, NextEra personnel did not properly implement procedure MS0519.17, "Crosby Pressurizer Mechanical Safety Valve Removal and Installation," that required the users of the safety valve bolting tension system to verify a gap formed between the piston and the locking nut on each bolt after tensioning. This resulted in eight bolts not being properly torqued during reinstallation of pressurizer safety valve RC-V-116, which led to reactor coolant system leakage past the valve's flange gasket that caused a valve high discharge temperature alarm on October 21, and required operators to return the plant to cold shutdown. Because this issue is of very low safety significance (Green) and NextEra entered this into their corrective action program as AR1815307 and implemented immediate corrective actions to retorque the bolts and replace the gasket on RC-V-116, this finding is being treated as an NCV consistent with the NRC Enforcement Policy. **(NCV 05000443/2012005-02, Failure to Adequately Implement Procedure Led to Reactor Coolant System Leakage from Pressurizer Safety Valve Flange)**

1R20 Refueling and Other Outage Activities (71111.20 – 1 sample)

a. Inspection Scope

The inspectors reviewed the station's work schedule and outage risk plan for the maintenance and refueling outage (OR15), which was conducted September 15, 2012, through October 31, 2012. The inspectors reviewed NextEra's implementation of outage plans and schedules to verify that risk, industry experience, previous site-specific problems, and defense-in-depth were considered. During the outage, the inspectors observed portions of the start-up processes and monitored controls associated with the following outage activities:

- Configuration management, including maintenance of defense-in-depth, commensurate with the outage plan for the key safety functions and compliance with the applicable technical specifications when taking equipment out of service
- Implementation of clearance activities and confirmation that tags were properly hung and that equipment was appropriately configured to safely support the associated work or testing
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and instrument error accounting
- Status and configuration of electrical systems and switchyard activities to ensure that technical specifications were met
- Monitoring of decay heat removal operations
- Impact of outage work on the ability of the operators to operate the spent fuel pool cooling system
- Reactor water inventory controls, including flow paths, configurations, alternative means for inventory additions, and controls to prevent inventory loss
- Activities that could affect reactivity
- Refueling activities, including fuel handling and fuel receipt inspections
- Fatigue management
- Identification and resolution of problems related to refueling outage activities
- Reactor criticality and main turbine generator start-up

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 4 samples)

a. Inspection Scope

The inspectors observed performance of surveillance tests and/or reviewed test data of selected risk-significant SSCs to assess whether test results satisfied technical specifications, the UFSAR, and NextEra procedure requirements. The inspectors verified that test acceptance criteria were clear, tests demonstrated operational readiness and were consistent with design documentation, test instrumentation had current calibrations and the range and accuracy for the application, tests were performed as written, and applicable test prerequisites were satisfied. Upon test completion, the inspectors considered whether the test results supported that equipment was capable of performing the required safety functions. The inspectors reviewed the following surveillance tests:

- Diesel generator 1A 18 month operability surveillance on October 13
- Control rod drive mechanism testing on October 26
- Turbine-driven emergency feedwater pump 18 month surveillance on October 26
- Reactor coolant system unidentified leakage monitoring calculation surveillance from December 24-27 (RCS)

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

During October 9-12, 2012, the inspectors reviewed and assessed NextEra's performance in assessing the radiological hazards in the workplace associated with Seabrook Station's Refueling Outage 15. The inspectors interviewed the Radiation Protection Manager, radiation protection supervisors, and radiation workers. The inspectors performed walkdowns of various portions of the plant, performed independent radiation dose rate measurements, observed work activities in Radiological Control Areas and reviewed relevant documents. The inspectors used the requirements in 10 CFR Part 20 and guidance in Regulatory Guide (RG) 8.38, "Control of Access to High and Very High Radiation Areas for Nuclear Plants," the technical specifications, and NextEra's procedures required by technical specifications as criteria for determining compliance.

.1 Radiological Hazard Assessment

The inspectors conducted walkdowns to evaluate material, work and radiological conditions in the containment, auxiliary building, and turbine building. The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation.

- Reactor assembly
- Cavity decontamination
- Primary steam generator inspections
- In-service inspections
- Scaffolding

For these work activities, the inspectors assessed whether the pre-work surveys performed were appropriate to identify and quantify the radiological hazard and to establish adequate protective measures. The inspectors evaluated the radiological survey program to determine if radiological hazards were properly identified, including: discrete radioactive hot particles, alpha emitter contamination; transuranics and hard to detect nuclides in air samples; transient dose rates; and large gradients in radiation dose rates.

The inspectors observed work in potential airborne areas and evaluated whether the air samples from the primary side of the steam generator and cavity during decontamination were representative of the breathing air zone and were properly evaluated. The inspectors evaluated whether continuous air monitors (CAMs) on the refueling floor and in containment were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors evaluated NextEra's program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

.2 Instructions to Workers

The inspectors reviewed the following radiation work permits (RWP) used to access high radiation areas (HRA) and evaluated if the specified work control instructions and control barriers were consistent with technical specification requirements for locked high radiation areas (LHRA).

- RWP 12-0031, Rx Cavity Work during OR15, including Rx Head Lift
- RWP 12-0038, Full Channel Head Entry: Install and Remove Nozzle Dams, Clean and Inspect Bowl
- RWP 12-0042, ISI Activities
- RWP 12-0043, Valve Work to Include Relief Valve Bench Testing
- RWP 12-0071, SI-V-82 Replacement Support Activities: No grinding, welding or abrasive activities allowed

For these radiation work permits, the inspectors assessed whether allowable stay times or permissible dose for radiologically significant work under each radiation work permit, were clearly identified. The inspectors evaluated whether electronic personal dosimeter alarm setpoints were in conformance with survey indications and plant procedural requirements.

The inspectors reviewed occurrences where a worker's electronic personal dosimeter noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the corrective action program and whether compensatory dose evaluations were conducted as appropriate.

For work activities that could suddenly and severely increase radiological conditions, i.e., fuel transfers from Containment into the Spent Fuel Building, the inspectors assessed NextEra's means to inform workers of these changes that could significantly impact their occupational dose.

.3 Radiological Hazards Control and Work Coverage

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients, e.g. inside the primary steam generators.

The inspectors reviewed the following radiation work permits for work within airborne radioactivity areas with the potential for individual worker internal exposures.

- RWP 12-0031, Rx Cavity Work during OR15, including Reactor Head Lift
- RWP 12-0038, Full Channel Head Entry: Install and Remove Nozzle Dams, Clean and Inspect Bowl

For these radiation work permits, the inspectors evaluated airborne radioactivity controls and monitoring, including the potential for significant airborne levels. The inspectors assessed applicable containment barrier integrity and the operation of temporary high-efficiency particulate air ventilation systems.

b. Findings

No findings were identified.

2RS2 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

The inspectors assessed performance with respect to maintaining occupational individual and collective radiation exposures as low as is reasonably achievable (ALARA) during Seabrook Station's Refueling Outage 15. The inspectors used the requirements in 10 CFR Part 20, the guidance in RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Plants will be As Low As Is Reasonably Achievable", RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposure As Low as Reasonably Achievable," technical specifications, and NextEra's procedures required by technical specifications as criteria for determining compliance.

.1 Radiological Work Planning

The inspectors selected the following work activities that had the highest radiological risk significance.

- ALARA Package (AP) No. 12-02, OR 15 Steam Generator Eddy Current Testing/Tube Plugging
- AP No. 12-01, Reactor Vessel Disassembly & Reassembly
- AP No. 12-04, OR 15 In-service Inspections
- AP No. 12-13, Cut Out and Replace SI-V-82
- AP No. 12-03, OR 15 Steam Generator Secondary Maintenance
- AP No. 12-05, OR 15 Cavity Decontamination
- AP No. 12-11, OR 15 Scaffolding

The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure reduction requirements. The inspectors determined whether NextEra reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The inspectors assessed whether NextEra's planning identified appropriate dose reduction techniques; considered alternate dose reduction features; and estimated reasonable dose goals. The inspectors evaluated whether NextEra's ALARA assessment had taken into account decreased worker efficiency from use of respiratory protective devices and/or heat stress mitigation equipment. The inspectors determined whether NextEra's work planning considered the use of remote technologies as a means to reduce dose and the use of dose reduction insights from industry operating experience and plant-specific lessons learned. The inspectors assessed the integration of ALARA requirements into work procedure and radiation work permit documents.

.2 Verification of Dose Estimates and Exposure Tracking Systems

The inspectors reviewed the assumptions and basis for the current collective exposure estimate for the outage and department dose for the year for accuracy. The inspectors reviewed applicable procedures to determine the methodology for estimating exposures from specific work activities and for department and station dose goals.

The inspectors evaluated whether NextEra had established measures to track, trend, and if necessary, to reduce occupational doses for ongoing work activities. The inspectors assessed whether dose threshold criteria were established to prompt additional reviews and/or additional ALARA planning and controls.

The inspectors evaluated NextEra's method of adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered. The inspectors assessed whether adjustments to exposure estimates were based on sound radiation protection and ALARA principles or if they were just adjusted to account for failures to plan/control the work.

.3 Radiation Worker Performance

The inspectors observed the performance of radiation workers and radiation protection technicians during refueling outage activities in radiation areas, airborne radioactivity areas, and high radiation areas. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice and whether there were any procedure compliance issues.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment (71124.04 - 1 sample)

a. Inspection Scope

The inspectors verified that occupational dose is being appropriately monitored and assessed during Seabrook Station's Refueling Outage 15. The inspectors used the requirements in 10 CFR Part 20, the guidance in RG 8.13 - Instructions Concerning Prenatal Radiation Exposures, RG 8.36 - Radiation Dose to Embryo Fetus, RG 8.40 - Methods for Measuring Effective Dose Equivalent from External Exposure, technical specifications, and NextEra's procedures required by technical specifications as criteria for determining compliance.

.1 External Dosimetry

The inspectors reviewed five corrective action program documents for adverse trends related to electronic personal dosimeters. The inspectors assessed whether NextEra had identified any adverse trends and implemented appropriate corrective actions.

.2 Internal Dosimetry

Routine Bioassay (In Vivo)

The inspectors reviewed procedures used to assess the dose from internally deposited radionuclides using whole body counting equipment. The inspectors evaluated whether the procedures addressed methods for differentiating between internal and external contamination, the release of contaminated individuals, determining the route of intake and the assignment of dose.

Dosimeter Placement and Assessment of Effective Dose Equivalent for External Exposures

The inspectors reviewed NextEra's methodology for monitoring external dose in non-uniform radiation fields or where large dose gradients exist. The inspectors evaluated NextEra's criteria for determining when alternate monitoring, such as use of multi-badging, is to be implemented.

The inspectors reviewed dose assessments performed for workers performing primary steam generator entries. These workers used multi-badge dosimetry that were used by NextEra to evaluate effective dose equivalent for this work activity, and the dose assessments performed were consistent with NextEra's procedures and dosimetric standards.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 – 3 samples)

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled NextEra's submittals for the Safety System Functional Failures performance indicator for the period of July 1, 2011, through November 30, 2012. To determine the accuracy of the performance indicator data reported during those periods, inspectors used definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, and NUREG-1022, “Event Reporting Guidelines 10 CFR 50.72 and 10 CFR 50.73." The inspectors reviewed NextEra’s licensee event reports (LER) to validate the accuracy of the submittals. The inspectors also reviewed the accuracy of the number of critical hours reported.

b. Findings

No findings were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

On December 20, 2012, the inspectors sampled NextEra’s submittals for the occupational radiological occurrences performance indicators (PI) for the period from the third quarter of 2011 through the third quarter of 2012. The inspectors used PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, dated October 2009, to determine the accuracy of the PI data that was reported for these four quarterly periods.

To assess the adequacy of the NextEra’s PI data collection and analyses, the inspector discussed the scope and breadth of its data review and the results of those reviews with the radiation protection staff. The inspector independently reviewed electronic personal dosimetry, accumulated dose alarms, occupational exposure reports, and dose assignments for any intakes that occurred during the time period reviewed to determine if there were any potentially unrecognized PI occurrences.

The inspector also reviewed conditions reports for occurrences related to radiation protection during the period. The inspector verified that no locked high and very high radiation area occurrences as defined in Seabrook’s technical specifications (TS) and NEI 99-02 were documented.

b. Findings

No findings were identified.

.3 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences

a. Inspection Scope

During December 20, the inspector sampled NextEra’s submittals for the radiological effluent TS/ODCM radiological effluent occurrences PI for the period from the third quarter 2011 through the third quarter 2012. The inspector used PI definitions and guidance contained in NEI 99-02 to determine the accuracy of the PI data that was reported for these four quarterly periods.

The inspector reviewed NextEra's corrective action 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. The inspector reviewed gaseous and liquid effluent summary data and the results of associated offsite dose calculations for selected dates between the third quarter of 2011 through third quarter of 2012, to determine if indicator results were accurately reported. The inspector also reviewed NextEra's methods for quantifying gaseous and liquid effluents and determining effluent dose.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152 – 2 samples)

.1 Routine Review of Problem Identification and Resolution Activities

a. Inspection Scope

As required by Inspection Procedure 71152, "Problem Identification and Resolution," the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that NextEra entered issues into the corrective action program at an appropriate threshold, gave adequate attention to timely corrective actions, and identified and addressed adverse trends. 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 corrective action program and periodically attended condition report screening meetings.

b. Findings

No findings were identified.

.2 Annual Sample: Review of 'Tin Whiskers' Related Component Failures

a. Inspection Scope

The inspectors performed an in-depth review of NextEra's evaluation and corrective actions associated with the failure of a solid state protection system (SSPS) input relay caused by a phenomenon known as 'Tin Whiskers'.

To determine whether NextEra was appropriately identifying, characterizing, and correcting problems associated with this issue, the inspectors assessed NextEra's: problem identification threshold; apparent cause analysis of the event; extent of condition reviews; and the prioritization, timeliness, and adequacy of corrective actions.

The inspectors reviewed NextEra's apparent cause evaluation (ACE) and associated documentation for the failure, interviewed operations and engineering personnel, conducted a trend review for related failures occurring subsequent to the initial event, conducted a review of previous similar failures to ensure appropriate characterization, and reviewed NextEra's corrective action process procedures and close out documentation.

b. Findings and Observations

No findings were identified.

On December 22, 2011, during the planned performance of the train A solid state protective system (SSPS) logic test, the K252A input relay on the positive flux rate trip portion of Nuclear Instrument (NI) channel II was found to have conservatively failed in the closed position. The failure resulted in a partial reactor trip condition (one input to the two out of four logic required to trip the reactor) and an intermittent flashing bistable alarm to be received in the main control room. NextEra personnel cleared the condition by cycling the relay, but the condition reoccurred on December 25 and December 29, 2011. NextEra replaced the K252A relay after the December 29th occurrence. NextEra's apparent cause evaluation (ACE) (AR 01719149) determined that the primary cause of the condition was a phenomenon known as 'Tin Whiskers' in which a microscopic tin filament developed and created a bridge across two contacts of the relay.

The inspectors determined that NextEra's evaluation of the events appropriately identified the root and contributing causes. Additionally, the inspectors determined that the immediate and long term corrective actions developed as a result of the ACE were effective, and adequate to correct the root and contributing causes and reasonably prevent recurrence. A review of related condition reports generated in the ten months subsequent to the event, engineering failure analysis reports, and interviews with personnel from the Operations and Engineering departments support this conclusion.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a semi-annual review of site issues, as required by Inspection Procedure 71152, "Problem Identification and Resolution," to identify trends that might indicate the existence of more significant safety issues. In this review, the inspectors included repetitive or closely-related issues that may have been documented by NextEra outside of the corrective action program, such as trend reports, major equipment problem lists, system health reports, maintenance rule assessments, and maintenance or corrective action program backlogs. The inspectors also reviewed NextEra's corrective action program database for the third and fourth quarters of 2012 to assess condition reports written in various subject areas (equipment problems, human performance issues, etc.), as well as individual issues identified during the NRC's daily condition report review (Section 4OA2.1). The inspectors reviewed the NextEra department quarterly trend reports for the second and third quarters of 2012 to verify that NextEra personnel were appropriately evaluating and trending adverse conditions in accordance with applicable procedures.

b. Findings and Observations

No findings were identified.

As a result of this inspection, it was identified that there was a significant increase in the number of anonymous condition reports written in 2012 over prior years. A large majority of those condition reports were not related to nuclear safety or safety conscious work environment. NextEra wrote condition report AR 01823739 and is performing a common cause evaluation.

The inspectors did not identify any trends that had not already been identified by NextEra. NextEra maintenance identified weaknesses in administrative procedure adherence in their second quarter trend report, and NextEra engineering identified weaknesses in corrective action program procedure compliance in their second quarter trend report. Additionally, NextEra identified a trend in missed procedure steps during refueling outage 15 and performed a common cause evaluation under AR 01808393. The inspectors also noted that procedure use and adherence was identified as the cross-cutting aspect for three of the eight non-cited violations (NCV) identified by the NRC at Seabrook this year. This information suggests a potential emerging trend in this area.

In the previous semi-annual trend review documented in IR 05000433/2012003, the inspectors noted a trend in NRC findings concerning inadequate 10 CFR 50.59 screenings. NextEra had also identified a fleetwide trend in 10 CFR 50.59 screenings. To address this trend NextEra developed and conducted training with engineering personnel on 10 CFR 50.59 screenings, performed a self-assessment of the 10 CFR 505.59 process and implemented interim corporate oversight of 10 CFR 50.59 reviews. The inspectors conducted specific reviews in this area to assess the progress of corrective actions in this area and did not identify any incorrect 10 CFR 50.59 screenings among the samples reviewed.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 – 5 samples)

.1 Plant Events

a. Inspection Scope

On October 16, 2012, at 1920 hours, Seabrook declared an Unusual Event (#48413) because an earthquake was felt by onsite personnel within the protected area and the site confirmed the earthquake with the Manchester, NH dispatch office. Seabrook was in a refueling outage in Mode 5 with no reactivity manipulations in progress and the A train of shutdown cooling was in service. NextEra conducted multiple walkdowns of the site with no significant anomalies noted. At 0149 hours on October 17, 2012, Seabrook terminated their Notice of Unusual Event.

The inspectors reviewed the Seabrook technical support center response to the event and the use of the applicable procedures by NextEra personnel. The inspectors performed independent walkdowns of control room instrument panels and switchgear and accompanied NextEra staff on walkdowns in the primary auxiliary building. In addition the inspectors discussed the status and function of the seismic monitors that were used to assess the occurrence and magnitude of the earthquake with NextEra staff to confirm that the impact of the ground motion experienced by the plant remained within its design basis.

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report (LER) 05000443/2012-001-00: Inadequate Testing of Certain Emergency Feedwater Actuation System Relays

On July 18, 2012, during an extent of condition review, NextEra personnel identified that the response time for starting and loading of the motor driven emergency feedwater pump (EFW) had not been adequately tested. The inspectors reviewed the LER and the associated condition report CR 1785593, and verified that NextEra staff's evaluation and corrective actions were adequate. The enforcement aspects of performance deficiencies associated with this event were documented in Seabrook IR 05000443/2012004 under Section 4OA7. No new issues of concern were identified as a result of this additional review, this LER is closed.

.3 (Closed) Licensee Event Report (LER) 05000443/2012-002-00: Inadequate Testing of Response Time for Reactor Trip Breakers

On September 25, 2012, during a design change review and extent of condition review for the issues associated with LER 05000443/2012-001-00, as discussed above, NextEra personnel identified that the response time for the shunt trip circuit of the reactor trip breaker system had not been adequately tested. The inspectors reviewed this LER and the associated condition report, CR 1806525, that addressed this issue and verified that NextEra's evaluation and corrective actions were adequate. The enforcement aspects of the performance deficiencies related to this event are discussed in Section 4OA7. This LER is closed.

.4 (Closed) Licensee Event Report (LER) 05000443/2012-003-00: Reactor Trip Due to Circuit Board Failure that Closes Feed Regulating Valve

LER 2012-003, dated November 8, 2012, reported an event that resulted in a valid actuation of the Reactor Protection System and met the reporting criteria of 10 CFR 50.73 (a)(2)(iv)(A). The reactor trip was determined to be caused by the failure of an internal component on a printed circuit board in the 7300 process control system that caused the feedwater regulating valve for the C Steam Generator to close. As a result, the water level in the C Steam Generator decreased to the low level trip setpoint and initiated an automatic reactor trip.

The inspectors reviewed operator and equipment response to the event, the accuracy of the LER and compliance with 10 CFR 50.73 reportability requirements. No performance deficiencies were identified. This LER is closed.

.5 (Closed) Licensee Event Report (LER) 05000443/2012-004-00: Manual Actuation of the Service Water Cooling Tower

On October 31, 2012, NextEra staff manually initiated a tower actuation that transferred the cooling water source for the A-train service water (SW) from the ocean to the cooling tower. This actuation was in response to increased differential pressure across the SW strainer, indicating fouling of the strainer from ocean debris. The tower actuation was conducted in accordance with station procedures, and operators took appropriate actions based on the indications received. SW was returned to the normal supply on November 1, 2012.

The inspectors reviewed operator and equipment response to the event, the accuracy of the LER and compliance with 10 CFR 50.73 reportability requirements. No performance deficiencies were identified. This LER is closed.

4OA5 Other Activities

.1 Temporary Instruction 2515/187 – Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

a. Inspection Scope

Inspectors verified that NextEra's walkdown packages for the A and B diesel generator building, A and B residual heat removal (RHR) vault, and the mechanical penetration room contained the elements as specified in NEI 12-07 walkdown guidance document.

The inspectors accompanied NextEra personnel on their walkdown of the A and B RHR vault and mechanical penetration room and verified that NextEra confirmed the following flood protection features:

- Visual inspection of the flood protection feature was performed if the flood protection feature was relevant. External visual inspection for indications of degradation that would prevent its credited function from being performed was performed
- Critical SSC dimensions were measured
- Available physical margin, where applicable, was determined
- Flood protection feature functionality was determined using either visual observation or by review of other documents

The inspectors independently performed their walkdown and verified that the following flood protection features were in place.

- A and B diesel generator building on November 14, 2012

The inspectors verified that noncompliances with current licensing requirements, and issues identified in accordance with the 10 CFR 50.54(f) letter, Item 2.g of Enclosure 4, were entered into the NextEra's corrective action program. In addition, issues identified in response to Item 2.g that could challenge risk significant equipment and NextEra's ability to mitigate the consequences will be subject to additional NRC evaluation.

b. Findings

No findings were identified.

.2 Temporary Instruction 2515/188 – Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

The inspectors accompanied NextEra engineers on their seismic walkdowns of B RHR vault on August 29, 2012, the B emergency diesel generator (EDG) room on August 29, 2012, B essential switchgear room on August 30, 2012, and the primary auxiliary building elevation 25' on August 31, 2012, and verified that NextEra confirmed that the following seismic features associated with B RHR pump, B EDG, 4 kV Bus E6, and primary component cooling train B temperature element CC-TE- were free of potential adverse seismic conditions:

- Anchorage was free of bent, broken, missing or loose hardware
- Anchorage was free of corrosion that is more than mild surface oxidation
- Anchorage was free of visible cracks in the concrete near the anchors
- Anchorage configuration was consistent with plant documentation
- SSCs will not be damaged from impact by nearby equipment or structures
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment
- Attached lines have adequate flexibility to avoid damage
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding)

The inspectors independently performed their walkdown and verified NextEra's conclusions for the following SSCs:

- Spent fuel pool heat exchanger in spent fuel storage building on September 26, 2012
- motor driven pump (FW-P-37B) in the EFW pump house on November 13, 2012

Observations made during the walkdown that could not be determined to be acceptable were entered into the NextEra's CAP for evaluation.

Additionally, inspectors verified that items that could allow the spent fuel pool to drain down rapidly were added to the seismic walkdown equipment list and these items were walked down by NextEra.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

On January 2, 2013, the inspectors presented the inspection results to Mr. Kevin Walsh, site vice president, and other members of the Seabrook Station staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

4OA7 Licensee-Identified Violation

The following violation of very low safety significance (Green) was identified by NextEra and is a violation of NRC requirements that meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

- Seabrook technical specification surveillance requirement 4.3.1.2, “Reactor Trip System Instrumentation,” requires that the reactor trip system features response time for each reactor trip function listed in Table 3.3-1 be verified to be within its limit at least once per 18 months. On September 25, 2012, NextEra identified that the full scope of response time testing for the reactor trip system function had not been completed since initial licensing because the implementing procedure did not verify the response time for both reactor trip methods. Testing had been completed on the under-voltage circuit, but no testing had been performed on the shunt trip circuit. The issue was determined to be a violation of Seabrook TS 6.7, “Procedures and Programs,” which requires that written procedures be established, implemented and maintained as recommended in RG 1.33, Revision 2, Appendix A, February 1978. RG 1.33, Appendix A, requires implementing procedures for each SR listed in TSs. Contrary to this requirement, since initial licensing, NextEra’s procedure for implementing TS SR 4.3.1.2 did not test the response time for reactor trip breaker function at least once per 18 months, which resulted in a violation of TS 3.1.2, “Reactor Trip System Instrumentation,” as described in LER 05000443/2012-002-00. The finding was associated with the Mitigating Systems cornerstone and was evaluated for significance using Exhibit 2 of IMC 0609, Appendix A. Since the finding was not a design or qualification deficiency, did not result in a loss of system safety function, did not result in loss of a single train for greater than its allowed outage time, and was not potentially risk significant due to external events, the finding was determined to be of very low safety significance (Green). The issue was entered into NextEra’s CAP as CR 1806525.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
K. Walsh, Site Vice President
T. Vehec, Plant General Manager
P. Allen, Senior Radiation Protection Technician
J. Ball, Senior Engineer
K. Barry, Unit Supervisor
R. Bickford, Nuclear Systems Operator
K. Boehl, Senior Radiation Protection Analyst
P. Brangiel, Principal Engineer
B. Brown, Engineering Supervisor
V. Brown, Senior Licensing Engineer
J. Cadwallader, Nuclear Systems Operator
R. Campo, Engineering Supervisor
A. Caramihalis, Shift Manager
M. Chapman, Senior Engineer
M. Chevalier, Radiation Protection Supervisor
M. Collins, Design Engineering Manager
J. Connolly, Site Engineering Director
S. Doody, Unit Supervisor
K. Douglas, Maintenance Director
D. Flahardy, Radiation Protection Manager
S. Fournier, Principal Engineer
M. Griffin, Senior I+C Technician
S. Hamel, NDE Level III
D. Hampton, Senior Radiation Protection Analyst
L. Hansen, Principal Engineer
G. Kilby, Principal Engineer
T. Knott, Design Engineer
G. Kotkowski, Design Engineering Supervisor
M. Leone, Operations Training Supervisor
E. Mallett, Unit Supervisor
J. Mayer, Principal Engineer
B. McAllister, Plant Engineer
S. Morrissey, Electrical Maintenance Department Head
M. O'Keefe, Licensing Manager
R. Parry, Engineering Supervisor
V. Pascucci, Nuclear Oversight Manager
E. Pigott, Assistant Operations Manager
D. Robinson, Chemistry Manager
R. Sampson, Mechanical Maintenance Department Head
M. Scannel, Senior Radiation Protection Analyst
T. Smith, Radiation Protection Supervisor
D. Snyder, Senior Engineer
J. Sweeney, Principal Engineer
E. Trump, Senior Engineer
T. Vassallo, Principal Engineer

T. Waechter, Operations Director
K. Whitney, Principal Engineer
D. Yates, Senior Engineer

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened/Closed

NCV 05000443/2012005-01	NCV	Failure to Correct a Condition Adverse to Quality for the L-5 FICI Connection
NCV 05000443/2012005-02	NCV	Failure to Adequately Implement Procedure Led to Reactor Coolant System Leakage from Pressurizer Safety Valve Flange

Opened

None

Closed

LER 05000443/2012-001-00	LER	Inadequate Testing of Certain Emergency Feedwater Actuation System Relays
LER 05000443/2012-002-00	LER	Inadequate Testing of Response Time for Reactor Trip Breakers
LER 05000443/2012-003-00	LER	Reactor Trip Due to Circuit Board Failure that Closes Feed Regulating Valve
LER 05000443/2012-004-00	LER	Manual Actuation of the Service Water Cooling Tower

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection
Procedures
ON0043.06, Fire Pump House System Steam and Building Heating, Revision 8
ON1023.11, Service Water Cooling Tower Heating Systems Operation, Revision 5
ON1490.06, Winter Readiness Surveillance, Revision 10
OP-AA-102-1002, Seasonal Readiness, Revision 0

Condition Reports
01805522, Track Outstanding Issues Related to 2012 Winter Readiness
01815795, VBS Conditions Noted
01806318, Seasonal Readiness Roll-Up and Challenge Board Not Performed
01807961, Winter Readiness Declared without Sufficient Basis
01807965, Seasonal Readiness Not Presently a Living Process
01807967, Winter Readiness Review Boards Not Used Proactively

01807970, Winter Readiness Cert Letter Missing Required Items
01807972, EQT Related to Seasonal Readiness not Coordinated
01807975, Informal Tracking of Winter Readiness Related Items

Maintenance Orders/Work Orders
40127225, Winter Readiness Surveillance

Section 1R04: Equipment Alignment
Procedures
OS1016.01, Service Water System Fill and Vent, Revision 16
OS1016.03, Service Water Train A Operation, Revision 14
OS1026.10, Form A, DG 1B Lube Oil System Lineup, Revision 12
OS1026.11, Form A, DG 1B Jacket Cooling Water System Lineup, Revision 9
OS1026.12, Form A DG 1B Starting Air System Lineup, Revision 12
OS1026.13, Form A, DG 1B Fuel Oil System Lineup, Revision 10
OS1036.01, Aligning the Emergency Feedwater System for Automatic Initiation, Revision 17
OX1426.19, Aligning DG 1B Controls for Auto Start, Revision 3

Condition Reports
01818493, Received VAS Alarm that Train A SW Strainer D/P was High

Miscellaneous
System Health Report dated 7/1/12 thru 9/30/12
USFAR Chapter 6.8, Emergency Feedwater System, Revision 14
USFAR Chapter 9.2, Auxiliary Systems – Water Systems, Revision 15

Drawings
PID-1-DG-B20463, Diesel Generator Lube Oil System Train B Detail, Revision 20
PID-1-DG-B20464, Diesel Generator Fuel Oil System Train B Detail, Revision 17
PID-1-DG-B20465, Diesel Generator Starting Air System Train B Detail, Revision 24
PID-1-DG-B20466, Diesel Generator Cooling Water System Train B Detail, Revision 21

Section 1R05: Fire Protection
Procedures
MS0517.33, Inspection of Appendix A Fire Dampers, Revision 1
SSFP Prefire Strategies, Revision 0

Condition Reports
*01834448, Appendix A Fire Dampers -61’ RHR Not Shown on Pre Fire Drawing

Drawings
DG-F-1B-B, Diesel Generator Building Oil Tank Rooms Train B – Room DG102, Elev (-)16’ – 0”, Revision 00
EFP-F-1-A, Emergency Feedwater Pump House 27’-0”, Revision 00

Section 1R06: Flood Protection Measures
Procedures
PEG-265, Cable Condition Monitoring Program, Revision 0

Maintenance Orders/Work Orders
WO 40157915, Low Voltage Electrical Manhole and Vault Inspections
WO 40164038, Low Voltage Electrical Manhole and Vault Inspections

Miscellaneous
Cable Testing Results Spreadsheet
Cables in Manholes Spreadsheet
Fleet Dewater Program Spreadsheet

Section 1R08: In-service Inspection
Procedures:
Seabrook Station Engineering Procedure ESO1-1-001, Revision 6, Eddy Current Inspection of Pre-Service and In-service Heat Exchanger Tubing, expires 6/14/14
Westinghouse Procedure MRS-2,4,2 GEN-35, Revision 15: Eddy Current Inspection of Preservice and Inservice Heat Exchanger Tubing
Seabrook Station Engineering Procedure ES1807.042, Revision 01; Feedwater Piping Thermal Fatigue Cracking Ultrasonic Examination
Seabrook Station Engineering Procedure ES1807.032, Revision 01; Inservice Inspection Procedure Primary Containment Section XI IWE Program
Seabrook Station Engineering Procedure ES1807.030, Revision 03; Nondestructive Examination (NDE) Personnel Certification Program
Seabrook Station Engineering Procedure ES03-01-21, Revision 03; Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds PDI-UT-1; Expires June 18, 2014
Seabrook Station Engineering Procedure ES03-01-22, Revision 03; Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds; Expires January 20, 2013
Seabrook Station Engineering Procedure ES10-01-38, Revision 01; Manual Ultrasonic Procedure for the Examination of Non-PDI Nozzle Inner Corner Regions; Expires June 18, 2014
Seabrook Station Engineering Procedure ES10-01-39, Revision 01; Manual Ultrasonic Procedure for the Examination of Non-RPV Nozzle-to-Shell Welds in Vessels > 2”; Expires June 18, 2014
Seabrook Station Engineering Procedure ES1807.002, Revision 09; Liquid Penetrant Examination – Solvent Removable
Seabrook Station Engineering Procedure ES1807.012, Revision 06; Ultrasonic Thickness Measurements
Seabrook Station Engineering Procedure ES1807.025, Revision 05; Inservice Inspection (ISI) Visual Examination Procedure
Seabrook Station Mechanical Maintenance Procedure MS0517.43, Revision 02; Piping Installation And Maintenance
Seabrook Station Mechanical Maintenance Procedure MS0517.05, Revision 02, Change 02; Installation Of Pipe Supports
Seabrook Station Mechanical Maintenance Procedure MS0518.08, Revision 03, Change 01; Pipe Support Spring Can Setting And System Balancing
Seabrook Station Engineering Procedure ES0815.002, Revision 02; General Welding Procedure
Seabrook Station Engineering Procedure ES0815.003, Revision 01; Welding of Stainless Steel (P-8) Base Metals
Seabrook Station Administrative Procedure SH 6.7, Revision 04; Hexavalent Chromium Exposure Control
Seabrook Station Engineering Procedure ES1807.025, Revision 05; Inservice Inspection (ISI) Visual Examination Procedure

Westinghouse Procedure; Seabrook Appendix H & I Techniques Fall 2012 Inspection, 9/11/2012
Seabrook Station Engineering Procedure EX1803.001, Revision 05; “Complex Procedure” Reactor Containment Integrated Leakage Rate Test – Type A
Seabrook Station Mechanical Maintenance Procedure MS0526.09, Revision 05; On Stream Leak Repairs
Seabrook Station Administrative Procedure MA 10.3, Revision 08; Boric Acid Corrosion Control Program
Seabrook Station Engineering Procedure NACD, Revision 50; Temporary Modifications and Alterations (TMD/TMR)
Westinghouse Procedure, Seabrook Appendix H & I – 2012, MRS-TRC-2163 Revision 1, Seabrook Appendix H & I Techniques Fall 2012 Inspection**,** September 2012
Seabrook Station Mechanical Maintenance Procedure Westinghouse Swing Check Valve Maintenance, MS0519.44, Revision 07
Seabrook Station Engineering Procedure ES99-01-03, Revision 06, 1/14/13; Mechanical Ribbed Plugging of Steam Generator Tubes, LIMITED USE – SPECIAL CONDITIONS APPLY
Westinghouse Procedure MRS 2.3.2 GEN-13, Revision 26, 1/3/11; Mechanical Ribbed Plugging of Steam Generator Tubes
Seabrook Station Engineering Procedure ES12-01-30, Revision 00, 9/13/14; Steam Generator Tube Procedure Specification for Expanded Ribbed Mechanical Plugs
Westinghouse Procedure MRS-GEN-1196, Revision 16, 3/3/11; Steam Generator Tube Plugging Procedure Specification for Expanded Ribbed Mechanical Plugs
Seabrook Station Engineering Procedure ES01-1-100, Revision 06, 6/14/14; Eddy Current Inspection of Pre-Service and In-Service Heat Exchanger Tubing, 8/11/11
Westinghouse Procedure MRS 2.4.2 GEN-35, Revision 15, 6/22/11; Eddy Current Inspection of Preservice and Inservice Heat Exchanger Tubing, 8/11/11
Seabrook Station Engineering Procedure ES02-1-101, Revision 06, 7/20/14; Steam Generator Data Management
Westinghouse Procedure SGMS 2.2.1 GEN-011, Revision 17, 6/22/12; Steam Generator Data Management
Westinghouse Procedure MRS GEN-1127, Revision 11, 3/5/12; Guidelines for Steam Generator Eddy Current Data Quality Requirements
Seabrook Station Engineering Procedure ES03-01-100, Revision 03, 6/14/14; Guidelines for Steam Generator Eddy Current Data Quality Requirements

Section XI Repair/Replacement Samples:
EC 0000273936 003, 10/18/11; Installation of Westinghouse Leak Seal on valve 1SI-V-82
Form NIS-2A Repair/Replacement Certification Record, Plan Number 40086371-09; Replacement of valve SI-V-82, 3/19/12
Westinghouse Seal Cap Installation, Form NIS-2A REPAIR/REPLACEMENT CERTIFICATION RECORD, 3/19/2012
WO 40156138, OR15 CUTOUT AND REPLACE 1-SI-V-82 per Design Change FP52914
WO 40173674, SW 32” Common Discharge Tee Repair in OR15
WO 40189255, Perform Weld Repair At 32” Tee Weld Joint, Ref. EC277580, 10/20/12

Condition Reports:
01633034, Crack Discovered at the PMCAP Near SW-V-20
01636130, UT Results of SW Piping in PAB 53’ Indicate Thinning
01638595, WEKO Seal Leaked during Testing
01639537, UT Results for Elbow off SW-V-67 Indicates Wall Thinning
01643948, Weld Repairs for the FAC Project have Failed NDE Exam

01652573, Body to Bonnet Leak SI-V-82
01652598, SI-V-82 Steam Leak during VT2 Check of CTB
01665187, SIIR – Inservice Inspection Reference
01672599, ISI Relief Requests Needed for OR15
01690252, Pre-Outage Milestone 5A Will Not Be Met for OR15
01692829, OR 15 Milestone Not Met – Submittal of RI ISI Relief Request
01695388, Inspection of Pipe Downstream of SW-FO-8292 – Rust, Liner
01695975, Degraded Pipe Liner Degraded and ID Corrosion
01696001, Pipe Replacement Required due to Corrosion
01696060, Pipe Liner Degraded and Pipe ID Corrosion
01696083, Pipe Wall Loss Found under Liner Material
01707284, ISI Isometric Drawings Need Revision for OR15
01721363, HSB Global Standards (ANII) ASME AAI-1 Audit
01746134, UT Needed for 1-SW-V-515 Flange through Wall Leak
01755780, Service Water Pipe Support Floor Embedment Plate Corrosion
01775656, OE Action Required, ISI NDE
01808243, Eddy Current Probe Became Detached in the Hot Leg of the B Steam Generator
01803965, Extent of Condition PT's on SG Blowdown Isolation Valves
*01809517, Containment Liner IWE Exam Results Require Evaluation
01809556, SG B Single Axial Crack Indication
*01809561, Leak Chase System Configuration

*Denotes this report was generated as a result of this inspection.

<u>Drawings & Sketches:</u>
NextEra Energy Seabrook drawing EC273936, Revision 0, 8/12/11; 1-SI-V-82 Seal Cap
NextEra Energy Seabrook drawing 1-SI-D20446, Revision 16, 10/26/10; SAFETY INJECTION SYSTEM INTERMEDIATE HEAD INJECTION SYSTEM DETAIL
NextEra Energy Seabrook drawing 9763-D-805000, Revision 0, 1/11/78; WELD-END PREP DETAILS – INTERFACE WITH WESTINGHOUSE WELD-END PREP VALVES, NOZZLES & EQUIPMENT
UNITED ENGINEERS DRAWING 5000-F-1382, Revision 9C; STANDARD WELD END PREPARATION DETAILS FOR PIPE
UNITED ENGINEERS DRAWING 9763 1-SI-251-13, Revision 7, 4/19/83; SAFETY INJECTION 1-SI-251-7
NextEra Energy Seabrook drawing PID-1-RC-D20843, Revision 15, 3/31/11; REACTOR COOLANT SYSTEM LOOP NO. 3
WESTINGHOUSE drawing D-04804-8374D48, Revision 13; SWING CHECK VALVE MOD.06001CS99000000
UNITED ENGINEERS DRAWING UNIT 1 9763-F-805122, Revision 11, 2/16/81; CONTAINMENT STRUCTURE PIPING ZONE 45B PLAN AT EL. -26'-0"

<u>Engineering Evaluations, Analyses, Calculations & Standards:</u>
Calculation C-S-1-24004, Revision 07, 9/13/2012; Seabrook Reactor Vessel Head Effective Degradation Years (EDY) & Re-Inspection Year (RIY)
TEAM INC. CLAMP ASSEMBLY RECONCILLATION OF STUD & NUT Material, FP 100630, 5/21/11
Calculation C-S-1-10158, Revision 01; 1-SI-V-82 INJECTION PRESSURE BOLTING EVALUATION, 5/20/11
NextEra Energy-Seabrook Root Cause Evaluation (CR Number 01652573, 5/17/11) Body-to-bonnet gasket joint for 1-SI-V-82

ASME IWE VT-3 EXAMINATION INDICATIONS CONTAINMENT DOME AR209442, 10/30/09
ASME IWE VT-1 EXAMINATION INDICATION CONTAINMENT DOME AR01809517, 10/4/12
ENGINEERING EVALUATION CONTAINMENT ILRT BOUNDARY, 26 OCTOBER 2012

Weld Records and Weld Process Guidance:
Weld Traveler for W.O. 40189255-03, SW-1815 (Weld Repair), 10/20/12

Evaluation/Screening of Boric Acid Leakage:
BORIC ACID CORROSION CONTROL EVALUATION, 1-SI-V-82, 6/10/11
BORIC ACID CORROSION CONTROL EVALUATION, 1-SI-V-82, 7/13/11
BORIC ACID CORROSION CONTROL EVALUATION, 1-SI-V-82, 5/18/11
BORIC ACID CORROSION EVALUATION (EDI 30560) RC AZIMUTH 158 OUTLET NOZZLE TO SAFE-END CONNECTION, 10/2009
BORIC ACID CORROSION EVALUATION (EDI 30560) 1-CS-V-836, NO DATE
BORIC ACID CORROSION CONTROL EVALUATION, BRS-V-393, 5/03/11
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01815351, RCS Leak – Seal Table Location L-5 Failure
01815787, Investigate and Correct Erratic Incore Detector Indications
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05-07755, Westinghouse Letter–Potential Tin Whiskers on Printed Circuit Board Components
00104745, This CR Is to Track the Review of NRC Info Notice 2005-25
00127629, ACR 98-1777 on 6/14/98, S/G 'C' Lo-Lo Level Bistable Light
00167724, On 01/14/02 UL Status Light UL1-G/9 "SG D Lo PSR SI/MSI"
01718841, Ni-N-42 Positive Rate Alarm Cycling on MCB UL-6
01719149, UL-6 K-14, PWR Range 42B High Rate Trip Is Flashing
01720010, NI-42 High Rate Trip-Repeat Issue-D5015 DFS
01820401, Received D4874 and D4457 Alarms
01823501, Spurious Alarms Associated with SSPS
01827306, CR Received A Train SI Block Alarm for <P11 Blocks
01803271, Reactor Trip
01807950, MRC Identified Adverse Trend for Missed QC Inspection Holds
01808393, OR15 Trend for Workers Not Following Work Instructions

Miscellaneous
Information Notice 2005-25, Inadvertent Reactor Trip and Partial Safety Actuation Due to Tin Whiskers
LER 2012-003-00, Reactor Trip Due to Circuit Board Failure
Westinghouse Technical Bulletin TB-05-4, Potential Tin Whiskers on Printed Circuit Board Components
Operations Quarterly Trend Report, 3Q 2012
Security Quarterly Trend Report, 3Q 2012
Training Quarterly Trend Report, 3Q 2012
Engineering Department Quarterly Trend Report, 2Q 2012
Maintenance Quarterly Trend Report, 2Q 2012

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion
Procedures
D5452 Alarm Response Sheet
ES1802.001, Earthquake Response, Revision 01
IX1670.910, SM-X-6700, SM-X-6701, and SM-X-6710 Seismic Monitors Calibration, Revision 08
NM 11800, Hazardous Condition Response and Recovery Plan, Revision 24

Condition Reports
01813799, Unusual Event Declared per HU1 Natural Phenomena

Maintenance Orders/Work Orders
WO 40093785, Containment Foundation Seismic Monitor

Section 4OA5: Other Activities
Condition Reports
01813880, Fukushima Flooding Walkdown Small Available Physical Margin
01820679, Fukushima Flooding Design Basis Walkdown-Evaluation Required

Miscellaneous
EPRI-TR-1025286, Seismic Walkdown Guidance, Revision 7
NEI 12-07, Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features, Revision 0-A
Seismic Walkdown Equipment List for the Requirement 2.3 Walkdown, dated September 2012, Revision 1
UFSAR Chapter 3.4, Design of Structures, Components, Equipment and Systems – Water Level (Flood) Design, Revision 12

LIST OF ACRONYMS

ACE	Apparent Cause Evaluation
ADAMS	Agencywide Documents Access Management System
ALARA	As Low as is Reasonably Achievable
AR	Action Request
ASME	American Society of Mechanical Engineers
AVB	Anti-Vibration Bar
CAM	Continuous Air Monitor
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CR	Condition Report
ECT	Eddy Current Testing
EDG	Emergency Diesel Generator
EFW	Emergency Feed Water
EPRI	Electric Power Research institute
ESF	Engineered Safety Features
HRA	High Radiation Area
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LOCA	Loss of Coolant Accident
MR	Maintenance Rule
MT	Magnetic Particle Testing
NCV	Non-cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NI	Nuclear Instrument
RHR	Residual Heal Removal
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual

PI	Performance Indicator
PM	Preventive Maintenance
PT	Dye Penetrant Testing
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RG	Regulatory Guide
RP	Radiation Protection
RT	Radiographic Test (Radiography)
RWP	Radiation Work Permit
SAI	Single Axial Indication
SDP	Significance Determination Process
SG	Steam Generator
SSC	System, Structure or Component
SSPS	Solid State Protection System
SW	Service Water
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing/Examination
VHRA	Very High Radiation Area
VT	Visual Examination
WO	Work Orders