UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PA 19406-1415

August 12, 2011

Mr. Paul Freeman
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 05000443/2011003

Dear Mr. Freeman:

On June 30, 2011, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at Seabrook Station, Unit No. 1. The enclosed report documents the inspection findings discussed on July 13, 2011, with Mr. E. Metcalf and other members of your staff.

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

The report documents three NRC-identified findings of very low significance (Green) that were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the findings as non-cited violations (NCV) consistent with Section 2.3.2.a of the NRC Enforcement Policy.

If you contest any NCV 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 the Seabrook Station. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspector at the Seabrook Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any), will be available electronically for public inspection in the

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Sincerely,

Arthur L. Burritt, Chief
Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report No. 05000443/2011003
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Sincerely,

/RA/
Arthur L. Burritt, Chief
Projects Branch 3
Division of Reactor Projects

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-443
License No.:	NPF-86
Report No.:	05000443/2011003
Licensee:	NextEra Energy Seabrook, LLC
Facility:	Seabrook Station, Unit No.1
Location:	Seabrook, New Hampshire 03874
Dates:	April 1, 2011 through June 30, 2011
Inspectors:	W. Raymond, Senior Resident Inspector J. Johnson, Resident Inspector T. Moslak, Health Physicist A. Turilin, Project Engineer J. DeBoer, Reactor Engineer T. Burns, Reactor Inspector
Approved by:	Arthur Burritt, Chief Projects Branch 3 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000443/2011003; 04/01/2011-06/30/2011; Seabrook Station, Unit No. 1; Routine Integrated Report; Fire Protection; Operability Evaluations.

The report covered a three-month period of inspection by resident and regional specialist inspectors. Three Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross cutting aspect of a finding is determined using the guidance in IMC 0310, "Components Within the 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: Mitigating Systems

Green. The inspectors identified a non-cited violation (NCV) of Technical Specification (TS) 6.7.1.h, which requires that written procedures be established and implemented for the fire protection program. Contrary to TS 6.7.1.f, the inspectors identified combustible materials which were not controlled per fire protection procedure FP 2.2, Revision 12. Specifically, (i) combustible materials were stored within three feet of an energized sample panel in the primary auxiliary building room PB404, a PRA risk significant area; and, (ii) combustible materials in excess of the permissible amounts were stored in waste process building area WB505. The inspectors identified materials stored in WB505 in excess of FP 2.2 limits on three occasions. Collectively, the NRC observations indicate a weakness in the programmatic control of combustible materials despite the fact that in each case the combustible materials were promptly removed following identification by the inspector. Seabrook entered this performance deficiency into their corrective action program.

The performance deficiency was more than minor because, if left uncorrected, inadequate control of combustibles could affect the Mitigating Systems cornerstone objective to assure external factors (fires) do not impact the availability and reliability of systems which mitigate events. The inspectors assessed the finding using Appendix F of the Significance Determination Process (SDP). Based on a degradation rating of low, which screens to Green in the fire protection SDP, the finding is of very low safety significance. This finding has a cross-cutting aspect in Human Performance, Work Practices [H.4(b)] because Seabrook personnel did not follow procedures for the control of transient combustibles. (Section 1R05)

Green. The inspectors identified a non-cited violation (NCV) of Technical Specification (TS) 6.7.1.a that requires written procedures be established and implemented, including administrative procedures that define authorities and responsibilities for safe operation with respect to operability determinations. Contrary to TS 6.7.1.a, NextEra identified a degraded and nonconforming condition related to reduced modulus of elasticity for buildings housing safety related equipment on May 27, 2011 but did not complete an operability determination until EC250348 was issued on June 28, 2011 (AR1664399). The delayed entry of the issue into the corrective action process to assess operability was contrary to Section 4.3 of EN-AA-203-1001 that requires operability assessments be completed in a time frame commensurate with the safety significance of the issue (within 8 hours). Seabrook subsequently completed an evaluation of the concrete issues and determined that the buildings housing safety-related equipment remained operable. Seabrook entered this performance deficiency into their corrective action program.

The performance deficiency was more than minor because a reasonable doubt of operability for the affected concrete structures existed until further engineering evaluations were completed to demonstrate the structures and systems that they housed would remain functional under design and licensing basis conditions. The finding affected the Mitigating Systems cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events in order to prevent core damage. The issue was evaluated using IMC 0609, "Significance Determination Process" (SDP), and was determined to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not result in an actual loss of safety function, was not a loss of a barrier function, and was not potentially risk significant for external events. The finding had a cross cutting aspect in the area of problem identification and resolution, P.1(a), because NextEra did not enter identified degraded concrete conditions for several site buildings into the corrective actions process in a timely manner, which would have ensured the shift manager completed timely operability evaluations for the affected structures. (Section 1R15.3)

Green. The inspectors identified a non-cited violation (NCV) of Technical Specification (TS) 6.7.1.a that requires that written procedures be established and implemented, including administrative procedures that define authorities and responsibilities for safe operation with respect to operability determinations. Contrary to TS 6.7.1.a, NextEra identified a degraded condition related to service water flow to the B emergency diesel generator (EDG) heat exchanger (HX) on June 28, 2011 but did not fully evaluate the reduced flow under all plant conditions as required by NextEra procedure EN-AA-203-1001. Fouling of the heat exchanger tubes was subsequently identified and mitigated. Seabrook also completed an evaluation of the B EDG service water flow issues and determined that the EDG remained operable. Seabrook entered this performance deficiency into their corrective action program.

The performance deficiency was more than minor because a reasonable doubt of operability existed until further engineering evaluations were completed to demonstrate adequate service water flow to the B EDG HX existed and the B EDG remained functional under design and licensing basis conditions. The finding affected the Mitigating Systems cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events in order to prevent core damage. The issue was evaluated using IMC 0609, "Significance Determination Process" (SDP), and was determined to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not result in an actual loss of safety function, was not a loss of a barrier function, and was not potentially risk significant for external events. The finding had a cross cutting aspect in the area of problem identification and resolution, P.1(c), because NextEra personnel did not thoroughly assess EDG operability to assure reduced HX SW flow was acceptable under all operating conditions, or assure appropriate corrective actions were timely completed. (Section 1R15.4)

REPORT DETAILS

Summary of Plant Status

The plant was shutdown at the start of the report period to conduct refueling outage OR14. NextEra completed reactor refueling and maintenance activities on the reactor and plant secondary systems. The plant was started up and the reactor was taken critical on May 23, 2011, and operation at full power resumed on May 26, 2011. The turbine was taken offline for maintenance on the secondary plant on June 4, 2011. Seabrook returned to full power on June 6, 2011.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Preparation (71111.01 – 2 sample)

.1 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspector completed one seasonal extreme weather conditions inspection sample. The inspectors assessed NextEra's readiness for the onset of hot weather. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) descriptions for related design features and verified the adequacy of the station procedures for hot weather protection. The inspectors reviewed NextEra's actions per procedure ON1490.09 for seasonal readiness, and procedure OS1200.03 for severe weather. The inspectors also performed walkdowns of susceptible systems, specifically the emergency feedwater, electrical distribution and service water systems. The inspectors reviewed deficiencies related to extreme weather preparation and verified the issues were entered into the corrective action program. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2 Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

The inspectors completed one summer readiness of offsite and alternate AC power systems inspection sample. The review focused on NextEra procedure OS1246.02, "Degraded Vital AC Power." The inspectors verified that plant features were maintained and procedures for operation were adequate to ensure the continued availability of AC power systems. The inspectors verified that communication protocols with the transmission system operator were adequate to ensure that appropriate information was exchanged when issues arose that could impact the offsite power system. The inspectors also observed NextEra's implementation of OS1246.02 during periods that challenged grid conditions between April 1, 2011 and June 30, 2011. The inspection included walkdowns of the onsite normal and emergency AC power systems and the inspectors reviewed deficiencies related to summer readiness of offsite and alternate AC power systems and verified these issues were entered into the corrective action program. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04Q - 5 samples; 71111.04S – 1 sample)

.1 Partial Walkdown

a. Inspection Scope

The inspectors completed five partial system walkdown inspection samples for the plant systems listed below. The inspectors verified that valves, switches, and breakers were correctly aligned in accordance with Seabrook's procedures and that conditions that could affect system operability were appropriately addressed. The inspectors reviewed applicable piping and instrumentation drawings and system operational lineup procedures. The documents reviewed are listed in the Attachment.

- Primary component cooling water (PCCW) "A" Train with "B" service water (SW) and "B" PCCW out of service for work performed on April 11, 2011.
- "A" train primary component cooling water (PCCW) during planned unavailability of the "B" train PCCW and SW systems on April 27, 2011 through May 2, 2011.
- Reactor and support system alignments on April 22, 2011, in preparation for plant startup from Mode 6.
- Residual heat removal (RHR) system alignment for low temperature over pressure protection during shutdown cooling operations on April 1, 2011 through April 4, 2011.
- "B" train RHR on May 5, 2011 through May 10, 2011 during the removal and replacement of 1-RHR-P-8A motor and seal package.

b. Findings

No findings were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors completed one complete system walkdown inspection sample on the service water system, specifically, Train "A" during a Train "B" pipe replacement and SW ocean outage. The inspectors walked down the accessible portions of the system to verify the system's overall material condition; that valves were correctly positioned; that electrical power was available; that major system components were properly labeled; that hangers and supports were correctly installed and functional; and that ancillary equipment or debris did not interfere with system performance. The inspectors reviewed plant procedures, system drawings, the UFSAR, and the technical specifications (TS). The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05Q - 5 samples)

.1 Quarterly Review of Fire Areas

a. Inspection Scope

The inspectors completed five quarterly fire protection inspection samples. The inspectors examined the areas of the plant listed below to assess: the control of transient combustibles and ignition sources; the operational status and material condition of the fire detection, fire suppression, and manual fire fighting equipment; the material condition of the passive fire protection features; and the compensatory measures for out-of-service or degraded fire protection equipment. The inspectors verified that the fire areas were maintained in accordance with applicable portions of Fire Protection Pre-Fire Strategies and Fire Hazard Analysis. The documents reviewed are listed in the Attachment.

- Primary auxiliary building 53 FT (PAB-F-3A-Z).
- Containment 26 FT (C-F-3-2).
- Fuel storage building 7 FT (FSB-F-1-A).
- Site yard area with focus on containment outage access (PLT-F-1-0).
- Containment 0 FT and +25 FT (C-F-2-Z and C-F-3-Z).

b. Findings

1. Inadequate Control of Combustible Materials

Introduction: The inspectors identified a non-cited violation (NCV) of Technical Specification (TS) 6.7.1.h, which requires that written procedures be established and implemented for the fire protection program. The inspectors identified combustible materials that were not controlled per NextEra procedure FP 2.2 in the primary auxiliary building and in the waste process building room WB505. The inspectors identified materials stored in WB505 in excess of FP 2.2 limits on 3 occasions from April 15, 2011 to July 1, 2011. Collectively, the NRC observations indicate inadequate programmatic control of transient combustible materials. Seabrook removed the improperly stored material identified by the inspector and entered this performance deficiency into their corrective action program.

Description: Procedure FP 2.2, "Control of Combustible Materials", provides requirements for controlling combustible materials at Seabrook. The inspectors identified the following conditions that did not meet the requirements of FP 2.2:

(a) During a walkdown of the Waste Processing Building area WB505 on April 15, 2011, the inspectors identified ten rolls of new plastic bags. Section 4.7 of FP 2.2 allows permanent storage of combustible materials in room WB505 in quantities specified for normal operations on bag rack only (i.e., three rolls of bags). In addition to the bags on the rack, seven additional rolls of bags were beside the rack. Procedure FP 2.2, Section 4.4.3, provides a permit threshold for quantities that exceed 100 pounds of National Fire Protection Association (NFPA) flammability category 1 solid materials (Class A materials). No transient combustible material permit was issued. The additional seven rolls of bags exceeded 100 pounds. The inspectors discussed this issue with Operations Management (OM). The materials were removed.

On June 15, 2011, inspectors identified a similar condition in room WB505 in that three additional rolls of bags were near the bag rack. The inspectors discussed the issue with Shift Manager (SM) and Fire Brigade Leader (FBL). Condition Report 1661217 was initiated and the non-permitted materials were removed. On July 1, 2011, the inspectors identified the same conditions in room WB505. Specifically, three additional rolls of bags were near the bag rack. Further, there was a half-full 55 gallon drum of used oil in the area. No transient combustible permit existed for the materials. The inspectors discussed these observations with the on-duty FBL and SM. The non-permitted materials were removed. Condition Reports 1666354 and 1666363 were initiated to enter this issue in the corrective action program.

(b) During a walkdown of the primary auxiliary building (PAB) room PB404 on June 15, 2011, the inspectors observed a cardboard box of paper towels and partially filled plastic bag stored next to (within three feet) 1-SS-CP-166-B, an energized sample analysis control panel. The materials were used by chemistry personnel to obtain steam generator blowdown samples. The sample panel area is an elevated platform that is approximately three feet wide. Section 4.2.5.c of FP 2.2 states that materials are not to be stored within three feet of energized electrical equipment (panels, etc.). Room PB404 houses the primary component cooling water pumps and heat exchangers, and is a designated PRA risk significant area, defined as an area that contributes the greatest majority of risk of core damage to fire initiated events. The inspectors discussed the observations with the on-duty Fire Brigade Leader and Shift Manager. The combustible materials were removed. Condition Report AR 1661010 was initiated to enter this issue in the corrective action program.

Collectively, these NRC observations indicate programmatic weakness in the control of combustible materials. The failure by workers to follow procedures and the ineffective NextEra actions to keep combustibles in WB505 below limits set by FP 2.2 raise a concern with the control of combustibles which, if left uncorrected, could lead to a more significance safety concern. Further, the inspectors identified that restrictions contained in the Final Safety Analysis Report, Appendix A, Responses to BTP APCSB 9.5-1 - materials near safety related tanks, were not reflected in FP 2.2. NextEra entered this issue into the corrective action program as AR 1667113.

<u>Analysis</u>: The inspectors determined that the failure to properly implement procedure FP 2.2 for the control of transient combustible material was a performance deficiency. This finding was considered more than minor because, if left uncorrected, inadequate control of combustibles could affect the Mitigating Systems cornerstone objective to assure external factors (fires) do not impact the availability and reliability of systems which mitigate events.

The inspectors performed a significance determination of this issue using IMC 0609, "Significance Determination Process" (SDP), Appendix F, "Fire Determination Significance Determination Process."
The issue met the Phase I qualitative screening criteria as discussed in Appendix F. Based on an evaluation using Step 1 of Appendix F, the inspectors determined the finding affected the category of Fire Prevention and Administrative Controls in that combustible material was not being properly controlled; the finding had a "low" degradation rating; and, the finding was of very low safety significance (Green). The inspectors determined this event affected the cross-cutting area of H.4.(b), Human Performance, Work Practices, because of the failure of workers to follow station procedures.

Enforcement: TS 6.7.1.h requires that written procedures shall be implemented for the Fire Protection Program (FPP). Fire Program procedure FP 2.2, "Control of Combustible Materials," Revision12, limits the quantity of transient combustible material stored in WB505 (Section 4.7) and near electrical panels in PB404 (Section 4.2.5.c). Contrary to the above, NextEra did not limit the quantity of transient combustible material stored in WB505 and near electrical panels in PB404 in accordance with the requirements of procedure FP 2.2. Specifically, NextEra stored ten rolls of plastic bags in WB505 and stored a cardboard box of paper towels and a partially filled plastic bag within three feet of an energized sample analysis control panel in PB404. Because the failure to control combustible materials was of very low safety significance and has been entered into NexEra's corrective action program (ARs 1661010, 1666354, 1666363), this violation is being treated as an NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy **(NCV 05000443/2011003-01, Inadequate Control of Combustible Materials).**

1R07 Heat Sink Performance (71111.07 - 1 sample)

a. Inspection Scope

The inspectors completed one heat sink performance inspection sample. Specifically the inspectors reviewed the 2011 testing of the "B" component cooling water heat exchanger to verify that the heat exchanger could fulfill its design function. The inspectors reviewed thermal performance monitoring (WO 01202862), trending data for heat exchanger temperatures and fouling factors, and ES1850.017, "SW Heat Exchanger Program". The inspectors interviewed the system engineer to evaluate the process used to monitor the heat exchanger and commitments in Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The inspectors performed system walk downs and reviewed condition reports to verify that issues associated with the heat exchanger were identified and corrected. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R08 <u>Inservice Inspection</u> (71111.08 - 1 sample)

a. Inspection Scope

The purpose of this inspection was to review and assess the effectiveness of NextEra's In-service Inspection (ISI) program for monitoring degradation of the reactor coolant system (RCS) boundary, risk significant piping system boundaries, and the containment boundary. The inspectors reviewed a sample of nondestructive examination (NDE) activities to verify compliance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI and applicable NRC Regulatory Requirements. In addition, the inspectors reviewed samples of completed non-destructive examinations, inspection procedures and inspection test reports to verify compliance with the ASME Code, Section XI. The inspectors reviewed the results of the reactor vessel nozzle dissimilar weld evaluation in the post-MSIP configuration (AR1644106). Also, the inspectors reviewed repair and replacement activities which involved use of welding and NDE on pressure boundary risk significant systems.

The inspectors observed the performance of NDE activities in process and reviewed documentation and examination reports for additional nondestructive examinations. Non-destructive test processes inspected and reviewed included Visual (VT), Magnetic Particle (MT), Penetrant (PT), Eddy Current (ECT), and Ultrasonic (UT) testing. The sample selection was based on the inspection procedure objectives, risk significance and sample availability. The inspectors reviewed examination procedures, procedure and personnel qualifications and examination test results.

The inspectors reviewed the procedures used to perform visual examinations for indications of boric acid leaks from pressure retaining components including the vessel upper head penetrations and their connections to the control rod drive mechanisms. The inspectors reviewed samples of operability evaluations, engineering evaluations and corrective actions provided for active and inactive boric acid leaks and determined they were consistent with the requirements of the ASME Code and 10 CFR 50, Appendix B, Criterion XVI, Corrective Action.

The inspectors performed a visual examination of the containment steel shell at the zero and minus 26 foot elevations within containment to evaluate the reported condition of the liner coating. The inspectors reviewed a sample of test reports, photographs and condition reports initiated as a result of the liner inspection performed by NextEra. Corrective action specified for conditions identified were evaluated by the inspectors to assess that the engineering organization was involved in providing evaluation and disposition. The inspectors confirmed there was no notable damage or indication of leakage identified during the ASME Section XI Section IWE evaluation.

<u>Examinations Inspected:</u>

- Magnetic particle test (MT) of weld F0104, field weld integrally attached pre-engineered pipe cap to 24 inch carbon steel SW pipe. Work order 1198488, ASME XI, Code Class 3. MT examination procedure ES 1807.003 R 7 Ch 1.
- Liquid penetrant (PT) test of weld F0104, examination of root pass of carbon steel attachment weld of cap to pipe using work document 1198488, ASME XI, Code Class 3, PT examination procedure ES 1807.002 R 7 Ch 1.

- Ultrasonic thickness test (UT) of carbon steel SW piping to determine if wall thinning had occurred at various selected circumferential and axial locations of the pipe shown on drawing 1198488. The pipe wall thickness was measured using UT procedure ES 1807.012 R 5.
- Visual test (VT-2) of reactor pressure vessel bare metal upper head surfaces with attention to the area where control rod drives intersect the head using remote visual techniques using test procedure ES 10-01-23.

The inspectors reviewed the steam generator (SG) degradation assessment (DA) to determine that NextEra had reviewed and incorporated the results of the previous outage degradation assessment, operational assessment (OA) and condition monitoring (CM) assessment. The inspectors reviewed the eddy current test (ECT) procedure, sample plan and data acquired. It was noted that no steam generator tubes were identified which specified in-situ pressure testing or specified "plugging".

The inspectors reviewed documentation of rework/repair activities which specified the development of ASME Section XI repair plans with the use of welding processes to complete the repair. The work orders (WO) which detail these repair/replacement activities are:

- WO 1198488 02 repair of thru wall leak of SW pipe line SW 1814-1-156 and modify support 1814-SG-02 by installation of a pre-engineered pipe cap, Drawing SK-EC145189-2000. The Inspectors reviewed applicable welding and NDT procedures to determine compliance with the ASME XI Code requirements.
- WO 40055977 01 fabrication of new SW pipe spool and reducer for replacement of existing spool which was degraded (wall thinning). Pipe spool welds F0105, 0106 and 0107 were made using weld procedure ES0815.004. The inspectors reviewed the fabrication and inspection procedures to verify compliance with the ASME XI Code requirements.

The inspectors reviewed the replacement material, weld procedure specifications and qualifications, welder qualifications, weld filler metals, non-destructive tests acceptance criteria and post work testing for each activity. The documents reviewed are listed in the Attachment.

b. <u>Findings</u>

No findings were identified.

1R11 <u>Licensed Operator Requalification Program</u> (71111.11Q - 1 sample)

.1 <u>Quarterly Resident Inspector Review</u>

a. <u>Inspection Scope</u>

The inspectors completed one quarterly licensed operator requalification program inspection sample. Specifically, the inspectors observed simulator just-in-time training of licensed operators on April 28, 2011 for reactor and steam plant re-start activities. The inspectors observed formal classroom and simulator activities. The inspectors examined the operators capability to perform actions associated with high-risk activities, the

Emergency Plan, previous lessons learned items, and the correct use and implementation of procedures. The inspectors observed and reviewed the training evaluator's critique of operator performance and verified that deficiencies were adequately identified, discussed and entered into the corrective action program. The inspectors reviewed the simulator's physical fidelity in order to verify similarities between the Seabrook control room and the simulator. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q – 2 samples)

a. Inspection Scope

The inspectors completed two maintenance effectiveness inspection samples. The inspectors reviewed performance-based problems and completed performance and condition history reviews for the selected in-scope structures, systems or components (SSCs) listed below to assess the effectiveness of the maintenance program. Reviews focused on: proper Maintenance Rule (MR) scoping in accordance with 10 CFR 50.65; characterization of reliability issues; tracking system and component unavailability; 10 CFR 50.65 (a)(1) and (a)(2) classifications; identifying and addressing common cause failures, trending key parameters, and the appropriateness of performance criteria for SSCs classified (a)(2) as well as the adequacy of goals and corrective actions for SSCs classified (a)(1). For the periodic assessment inspection sample, the inspectors reviewed the assessment frequency, the performance criteria, the use of operating experience and corrective actions. The inspectors reviewed system health reports, maintenance backlogs, and MR basis documents. The documents reviewed are listed in the Attachment.

- Residual heat removal (RH) system classified as MR (a)(2) with a focus on component performance impacting unavailability and reliability (AR 1647943).
- SW system classified as MR (a)(2) with a focus on pipe wall thinning identification and repair (ARs 1612061, 1637922, 1639537).

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 6 samples)

a. Inspection Scope

The inspectors completed six maintenance risk assessment and emergent work control inspection samples. The inspectors reviewed the scheduling and control of planned and emergent work activities in order to evaluate the effect on plant risk. The inspectors conducted interviews with operators, risk analysts, maintenance technicians, and engineers to assess their knowledge of the risk associated with the work, and to ensure that other equipment was properly protected. The inspectors reviewed the availability of opposite train guarded and protected equipment. The compensatory measures were evaluated against Seabrook procedures, Maintenance Manual 4.14, "Troubleshooting," Revision 0 and Work Management Manual 10.1, "On-Line Maintenance," Revision 3. Specific risk assessments were performed using Seabrook's "Safety Monitor", as applicable. The inspectors reviewed the maintenance items listed below. The documents reviewed are listed in the Attachment.

- Risk mitigation actions for orange risk condition associated with reactor head removal on April 5, 2011 through April 6, 2011 (WO 1205089).
- Planned work associated with SW "A" Train during "B" train pipe replacement and maintenance WO 00626035 for work performed April 14, 2011.
- Risk mitigation actions due to unplanned entry into orange risk condition due to degraded grid during planned work to remove steam generator nozzle dams on April 22, 2011 (WO 1203031).
- Emergent work associated with the heavy lift for the removal and replacement of "A" train RHR pump (1-RH-P-8A) on May 6, 2011 and May 9, 2011 (WO 40083875).
- Emergent work associated with the "A" train SW piping replacement and maintenance for leak downstream of heat exchanger isolation valve SW-V-16 on April 20, 2011 through April 22, 2011 (WO 40078357).
- Emergent work associated with the temporary repair of safety injection system check valve SI-V82 which had a body to bonnet leak that was leak sealed with the plant at normal operating temperature and pressure on May 18, 2011 through May 22, 2011 (WO 40086371).

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15 – 5 samples)

a. Inspection Scope

The inspectors completed five operability evaluation inspection samples. The inspectors reviewed operability evaluations and condition reports to verify that identified conditions did not adversely affect safety system operability or overall plant safety. The evaluations were reviewed using criteria specified in NRC Regulatory Issue Summary 2005-20, "Revision to Guidance formerly contained in NRC Generic Letter 91-18, Information to Licensees Regarding two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability" and Inspection Manual Part 9900, "Operability Determinations and Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety." In addition, where a component was determined to be inoperable, the inspectors verified that TS limiting condition for operation implications were properly addressed. The documents reviewed are listed in the Attachment. The inspectors also performed field walk downs and interviewed personnel involved in identifying, evaluating or correcting the identified conditions. The following items were reviewed:

- AR1662418, operability of the pressurizer code safety relief valve (1-RC-V-117) due to seat leakage, June 20, 2011.

- AR1641413, evaluation of containment shell with craze cracking in concrete, April 20, 2011.
- AR1644074, operability of containment enclosure building with reduced modulus of elasticity, April 21, 2011.
- AR1664399, operability of concrete structures with reduced modulus of elasticity, June 27, 2011.
- AR1664708, operability of B diesel generator with cooling water flow oscillations, June 28, 2011

b. Findings

NextEra wrote ARs1644074 and 1664399 to document the preliminary laboratory results for concrete core samples taken for the containment enclosure building (CEB) and four other seismic Category I buildings. Twenty core samples were taken as part of an extent of condition investigation for AR 581434 in which NextEra determined that sections of the below grade concrete walls could be affected by alkali-silica reaction (ASR). Prior NRC review of this area was documented in Inspection Reports 2010-04, 2010-05, 2011-02 and 2011-07.

.1 Inadequate 50.59 Screening for Design Change EC 272057 – AR1664074

NextEra issued EC272057, "Concrete Modulus of Elasticity Evaluation," on April 24, 2011 to address the results of testing that showed a reduction in the concrete modulus of elasticity in the CEB (AR 1644074). EC272057 also address the reduced modulus in the Control Building/Electric Tunnel CB/ET (AR581434). The lowest measured modulus was 2.16E+03 ksi for the CEB and 2.1E+03 ksi for the CB/ET, both less than the design value of 3.62E+03 ksi. EC272057 was supported by calculations C-S-1-10150 and C-S-1-10156 which reflected the degraded conditions in the design calculations CD-20-CALC and CE-4-CALC for the control and containment enclosure buildings, respectively.

NextEra concluded the structures remained operable and used EC272057 to disposition the degraded condition as "use-as-is," by incorporating the degraded condition into the design basis. In a safety evaluation screen per 10 CFR 50.59 for EC272057, NextEra concluded the change to the facility did not require a complete evaluation per 50.59(c)(2) because adequate design margin existed and there was no adverse affect on an UFSAR described design function.

The inspectors determined the 50.59 Screen for EC272057 did not correctly address Screen Question 5.a: *"Does the proposed activity involve a change to an SSC that adversely affects an UFSAR design function?* Using the guidance of the Seabrook 10CFR5059 Resource Manual and NEI 96-07, Revision 1, the inspectors determined that a 50.59 evaluation is specified for changes that adversely affect design function. In this situation, the ASR impacted concrete with reduced modulus of elasticity which reduces the flexural capacity of the walls would be an adverse effect. Therefore, NextEra should have evaluated the change to the facility per 10 CFR 50.59(c)(2).

The item is unresolved pending action by NextEra to complete a full 50.59 evaluation for EC272057 and subsequent NRC review of that evaluation to determine whether the performance deficiency is more than minor. (**URI 05000443/2011003-02, Inadequate 50.59 Screening for Design Change EC 272057**).

.2 Effects of Reduced Modulus on Concrete Structures – AR1644074 and 1664399

NextEra's analysis of the CEB samples found that the concrete has acceptable compressive strength and reduced but acceptable modulus of elasticity. To evaluate the effects of the reduced modulus, NextEra assessed the increase in strain for CEB building elements and found that the strain at the most limiting element remained less than the American Concrete Institute ACI-318 design stress limit and thus was acceptable. NextEra evaluated the impact on flexural capacity by reviewing the change in bending moment of structural elements. The reduced modulus causes the concrete to have increased flexure which has the effect of shifting the balance point in how load is transferred between the concrete and the imbedded steel (rebar). The reduced modulus causes a shift toward the reinforced steel in tension. The resultant change in bending moment was evaluated to show that the reduction in capacity was minimal and the stresses on the steel and concrete remain below the design stress limits with margin. NextEra's evaluation of the condition concluded that a change in the dynamic seismic response of the structure would be minor, and the CEB remains capable of performing its design function.

The prompt operability determination for the CEB (ARs 1644074 and 1664399) evaluated how the reduced modulus would affect the structure by analysis of locally impacted sections. The evaluation did not address the effects of reduced modulus on the changes to the natural frequencies of the structure and the global response of the structure to seismic loads. The inspectors requested further information on the effects of the reduced modulus on stresses and strain in the concrete and rebar system for which NextEra will complete additional analyses.

The prompt operability determination (POD) for the Control Building (AR581434) as well as for other ASR impacted structures (AR1664399) evaluated the effects of reduced modulus on portions of the below grade structures and the components housed within them. The evaluations lacked details to explain the effects of the reduced modulus on structural flexure as related to components attached to the structures, such as pipe supports and cable trays. Similarly, the evaluations lacked details with regard to the structure's response to seismic events as related to structure rigidity and changes in the natural frequency, and the bases to use the ground response spectra. Further, the evaluations lacked details to explain how the function of support anchor bolts would not be adversely impacted by reduced concrete compressive strength in the CB/ET.

This item is unresolved pending further NRC review of the above issues, action by NextEra to complete additional analysis of the CEB conditions and subsequent NRC Region I review of that analysis, and the completion of reviews by the NRC Office of Nuclear Reactor Regulation specified in the associated task interface agreement (ADAMS No. ML111610530). The result of these reviews will determine whether there is a performance deficiency associated with this item. (**URI 05000443/2011003-03, Operability Evaluation for Degraded Concrete in ASR Affected Plant Structures**).

.3 Untimely Operability Determination – AR 1664399

Introduction. The inspectors identified a Green non-cited violation (NCV) of Technical Specification (TS) 6.7.1.a that requires written procedures be established and implemented, including administrative procedures that define authorities and responsibilities for safe operation. NextEra identified a degraded and nonconforming condition related to reduced modulus of elasticity for buildings housing safety related equipment on May 27, 2011, but did not complete an operability assessment until June 27, 2011, when AR1664399 and EC250348 were issued. The delayed entry of the issue into the corrective action process to assess operability was contrary to Section 4.3 of EN-AA-203-1001 that requires an operability determinations (OD) be completed in a time frame commensurate with the safety significance of the issue (in most cases within 8 hours).

Description. Procedure EN-AA-203-1001, "Operability Determinations/Functional Assessments," provides requirements for evaluation of degraded conditions and nonconforming conditions and requires: the Shift Manager (a licensed Senior Reactor Operator) make an OD for each condition that involves equipment issues related to the ability of an SSC to perform its TS function (Section 3.2.1); degraded or nonconforming conditions be entered in to the corrective action program (Section 4.2.1); an immediate OD be performed following the discovery of a degraded or nonconforming condition (Section 4.1.7); and, the immediate operability determination shall be completed in a manner commensurate with the safety significance (in most cases during the shift when a concern was generated or 8 hours) and consider all plant conditions (Section 4.3.1).

On April 21, 2011, NextEra issued AR 1644074 upon receipt of initial test results showing the modulus of elasticity for concrete core samples taken from the containment enclosure building (CEB) was below the American Concrete Institute ACI-318 design value 3.62 E+03 ksi. A measured modulus as low as 2.16E+03 ksi (60% of the design value) was a degraded and nonconforming condition with respect to the properties of concrete in Category I structures as described in Section 3.8 of the UFSAR. A reduced modulus impacts the flexural capacity of the impacted walls and thus the function of the building. The Shift Manager, with input from Engineering, documented the basis for an immediate operability determination for the CEB in AR 1644074. NextEra issued EC250348 and Calculation C-S-1-10156 on April 25, 2011, which evaluated the integrity of the CEB with consideration of the reduced modulus to disposition the CEB as operable. The evaluations supporting EC250348 relied upon Calculation C-S-110150, completed for the Control Building on September 23, 2010, to show that the reduced modulus had minimal impact on flexure and bending moment capacity of the building walls that are heavily reinforced with steel. NextEra also initiated core sampling in several buildings including – the equipment vault, the emergency feedwater the emergency diesel generator buildings as part of the extent of condition review for the issues identified in the control building.

The initial report for the results of the additional testing performed identified reduced modulus of elasticity in all of the buildings in the expanded scope (equipment vault, emergency feedwater and emergency diesel generator buildings). The information was provided to the responsible engineer and made available in a draft report to NextEra on May 27, 2011, but subject to further review and comment with the vendor for final acceptance. On June 27, 2011, NextEra issued AR1664399 with an immediate operability determination to address the same reduced modulus condition described in

AR1644074. A Prompt Operability Determination (POD) was issued on June 28, 2011, to disposition the degraded condition for all impacted buildings which determined the structures were fully operable with margins. The inspectors identified that, although the data for the 3 buildings was preliminary (final reports were not issued by the vendor until July 1 and 27), NextEra should have initiated a condition report on May 27, 2011 to establish an immediate operability determination for the buildings since the reduced concrete modulus was a degraded and nonconforming condition as described in UFSAR Section 3.8. The failure to initiate a timely operability determination on May 27 was contrary to Sections 4.2.1 and 4.3.1 of EN-AA-203-1001. The failure to promptly enter a degraded condition into the corrective actions process to allow the Shift Manager to make timely operability evaluations was a performance deficiency.

Analysis. The inspectors determined that the failure to properly implement procedure EN-AA-203-1001 for the degraded and nonconforming condition discussed above was a performance deficiency. This performance deficiency was considered more than minor based on a comparison with Examples 3.j and 3.k of Appendix E of IMC 0612. Specifically, the performance deficiency was more than minor because a reasonable doubt of operability for the affected concrete structures existed until further engineering evaluations were completed to demonstrate the structures and systems that they housed would remain functional under design and licensing basis conditions. The finding affected the Mitigating Systems cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events in order to prevent core damage. The issue was evaluated using IMC 0609, "Significance Determination Process" (SDP), and was determined to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not result in an actual loss of safety function, was not a loss of a barrier function, and was not potentially risk significant for external events. The finding had a cross cutting aspect in the area of problem identification and resolution, P.1(a), because NextEra did not enter identified degraded concrete conditions for several site buildings into the corrective actions process in a timely manner that would have ensured the shift manager completed timely operability evaluations for the affected structures.

Enforcement. Technical Specification 6.7.1.a, Procedures and Programs, requires that procedures be established and implemented covering administrative procedures that define authorities and responsibilities for safe operation. Procedure EN-AA-203-1001 defines responsibilities and requirements for making immediate ODs to establish the acceptability of continued plant operation when SSCs are found to be degraded or nonconforming. Contrary to the above, NextEra did not make an immediate operability determinations to establish the acceptability of continued plant operation when the reduced concrete modulus for several plant structures was identified on May 27, 2011. Because this failure to make timely operability determinations is of very low safety significance and was entered into NextEra's Corrective Action Program (CR1673102), this violation is being treated as an NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy. **(NCV 05000443/2011003-04, Untimely Operability Determination for Degraded Concrete Structures Housing Safety-Related Equipment)**

.4 Inadequate Operability Determination – AR 1664708

Introduction. The inspectors identified a Green non-cited violation (NCV) of Technical Specification (TS) 6.7.1.a that requires that written procedures be established and implemented, including administrative procedures that define authorities and responsibilities for safe operation. NextEra identified a degraded condition related to reduced service water (SW) flow to the B emergency diesel generator (EDG) heat exchanger (HX) on June 28, 2011, but did not fully evaluate the reduced flow under all plant conditions as required by NextEra procedure EN-AA-203-1001.

Description. Procedure EN-AA-203-1001, "Operability Determinations/Functional Assessments," provides requirements for evaluation of degraded conditions and nonconforming conditions and requires: the Shift Manager (a licensed Senior Reactor Operator) make an OD for each condition that involves equipment issues related to the ability of an SSC to perform its TS function (Section 3.2.1); degraded or nonconforming conditions be entered in to the corrective action program (Section 4.2.1); an immediate OD be performed following the discovery of a degraded or nonconforming condition (Section 4.1.7); and, the immediate operability determination shall be completed in a manner commensurate with the safety significance (in most cases during the shift when a concern was generated or 8 hours) and consider all plant conditions (Section 4.3.1).

On June 28, 2011, NextEra issued AR1664708 when operators observed reduced SW flow through the B EDG heat exchanger during weekly testing of valve SW-V18. The flow initially varied from 500 to 1000 gpm, but increased to 1400 gpm with additional valve strokes. The operators questioned the adequacy of flow indication from SW-FE-6191, but the operability evaluation documented in AR1664708 accepted the flow with "no operability issues noted" and with plans to monitor the condition. NextEra observed "normal" flow during a subsequent valve stroke.

The operators again observed reduced and erratic SW flows (800-1500 gpm) during the next valve test on July 9, 2011. The B EDG was declared inoperable, AR1667857 was written and a POD per Section 4.3.1.C of EN-AA-203-1001 was requested at that time. A more detailed investigation and evaluation determined that flow element SW-FE-6191 was partially plugged contributing to variability in flow indication (but not the reduction), and that marine growth inside the SW pipe (line 1806-1-153-16) upstream of diesel heat exchanger DG-E42B was causing intermittent fouling of the heat exchanger tubes and reduced flow. Corrective actions were initiated to address potential heat exchanger fouling in both EDG SW cooling loops. The engineering evaluation associated with the POD confirmed that the "B" EDG and cooling subsystem was and had been fully operable under design basis conditions, including cooling water temperatures at the environmental extremes for operation on the ocean or the cooling tower.

The inspectors noted that diesel operating procedure OS 1026.09 requires a minimum of 900 gpm when SW cooling is provided on the ocean, and 1800 gpm when SW cooling is provided on the cooling tower. The inspectors noted further that normal SW flow through the heat exchanger varies from 1500 to 1900 gpm depending on ocean level at the intake. The inspectors determined that the June 28 operability evaluation documented in AR 1664708 lacked sufficient basis to explain the reduced flow at 500-1400 gpm, and failed to fully evaluate the EDG cooling function. The June 28 evaluation did not fully evaluate flow under all plant conditions as specified by Section 4.3.1.A of EN-AA-203-1001, namely, whether flow was adequate relative to the 1800 gpm needed for operation on the cooling tower. NextEra should have fully investigated the flow anomaly on June 28 and evaluated EDG operability under all design basis conditions per Section 4.3.1.C of EN-AA-203-1001, since reduced SW flow in the range of 500 to 1000 gpm was a degraded condition that impacted engine cooling. When EDG cooling was further investigated on July 9, the presumption on June 28 that the flow anomaly was caused by erratic indication was proven wrong. The initial incomplete operability evaluation resulted in the delayed identification, assessment and mitigation of a degraded condition impacting the functionality of the EDG. The failure to promptly and thoroughly evaluate degraded conditions for operability was a performance deficiency.

Analysis: The inspectors determined that not properly implementing procedure EN-AA-203-1001 for the degraded condition discussed above was a performance deficiency. This performance deficiency was considered more than minor based on a comparison with Examples 3.j and 3.k of Appendix E of IMC 0612. Specifically, the performance deficiency was more than minor because a reasonable doubt of operability existed until further engineering evaluations were completed to demonstrate adequate service water flow to the B EDG HX existed and that the B EDG remained functional under design and licensing basis conditions. The finding affected the Mitigating Systems cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events in order to prevent core damage. The issue was evaluated using IMC 0609, "Significance Determination Process" (SDP), and was determined to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not result in an actual loss of safety function, was not a loss of a barrier function, and was not potentially risk significant for external events. The finding had a cross cutting aspect in the area of problem identification and resolution, P.1(c), because NextEra personnel did not adequately evaluate operability to ensure that EDG cooling flow was acceptable under all operating conditions and assure appropriate corrective actions were timely completed.

Enforcement. Technical Specification 6.7.1.a, Procedures and Programs, requires that procedures be established and implemented covering administrative procedures that define authorities and responsibilities for safe operation. Procedure EN-AA-203-1001 defines responsibilities and requirements for making immediate ODs to establish the acceptability of continued plant operation when SSCs are found to be degraded. Contrary to the above, NextEra did not fully evaluate degraded service water flow on June 28, 2011, resulting in the delayed identification, assessment and correction of a condition that impacted B EDG cooling. Because the finding is of very low safety significance and was entered into NextEra's corrective action program (CR1673102), this violation is being treated as an NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy. **(NCV 05000443/2011003-05, Inadequate Operability Determination for Reduced EDG HX Cooling Water Flow)**

1R18 Plant Modifications (71111.18 - 3 samples)

.1 Permanent Modification – EC 145280: Project 52 SY Upgrade

a. Inspection Scope

The inspectors completed one permanent modification inspection sample. The inspectors reviewed modification package EC145280 that completed changes in the 345 kV switchyard to enhance reliability. The modifications included the installation of new breakers and bus sections to connect the Seabrook generator and unit auxiliary transformer to the grid. The review was completed to verify that the design bases and performance capability of the system was not degraded. The inspectors verified the new configuration was accurately reflected in the design documentation, and that the post-modification testing was adequate to ensure the SSCs would function properly. The inspectors interviewed plant staff, and reviewed issues entered into the corrective action program to verify that NextEra was effective at identifying and resolving problems associated with temporary modifications. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2 Temporary Modification – EC 272290: Install Varistor in Panel for CBA-CP-177

a. Inspection Scope

The inspectors completed one temporary modification inspection sample. The inspectors reviewed modification package EC 272290 associated with operation of the A EDG. The modification installed a varistor in the unit sub panel for CBA-CP-177. The varistor was installed across relay coils 52X and 52Y to minimize electrical transients when the anti-pump and breaker closing relays operated. The purpose of the varister is to suppress induced voltages in the emergency power sequencer logic circuits that can negatively affect proper operation of the A emergency diesel generator (CR1645405). The review was completed to confirm that the design bases and performance capability of the system were not degraded. The inspectors verified the new configuration was accurately reflected in the design documentation (reference Drawing 310926 Sheet AC4b), and that the post-modification testing was adequate to ensure that affected SSCs would function properly. The inspectors also interviewed plant staff, and reviewed issues entered into the corrective action program to verify that NextEra was effective at identifying and resolving problems associated with temporary modifications. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.3 Temporary Modification – EC 272512: Leak Sealing of SI-V82

a. Inspection Scope

The inspectors completed one temporary modification inspection sample. The inspectors reviewed modification package EC 272512 that installed a mechanical clamp and seal on safety injection system check valve SI-V82. The leak seal was installed to minimize gasket leak at the body to bonnet flange identified during plant startup to begin operating cycle 15. The review was completed to confirm that the design bases and performance capability of the system were not degraded. The inspectors verified the new configuration was accurately reflected in the plant documentation and that the clamp installation and sealing process would not adversely affect the check valve design functions. The inspectors also interviewed plant staff, and reviewed issues entered into the corrective action program to verify that NextEra was effective at identifying and resolving problems associated with temporary modifications. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 – 7 samples)

a. Inspection Scope

The inspectors completed seven post-maintenance testing (PMT) inspection samples. The inspectors observed portions of PMT activities in the field to verify the tests were performed in accordance with the approved procedures. The inspectors assessed the test adequacy by comparing the test methodology to the scope of the maintenance work performed. The inspectors evaluated the test acceptance criteria to verify that the test procedure ensured that the affected systems and components satisfied applicable design, licensing bases and TS requirements. The inspectors also reviewed recorded test data to confirm all acceptance criteria were satisfied during testing. The documents reviewed are listed in the Attachment. The activities reviewed are listed below:

- Retest of main steam to emergency feedwater pump turbine steam supply valve 1-MS-V-393 on May 18, 2011, following overhaul per WO 40065448.
- Retest of chemical and volume control system charging flow control valve 1-CS-FCV-121 on May 18, 2011, following overhaul per WO 1199620.
- Retest of "B" train charging pump 1-CS-P-2B on April 26, 2011, following motor replacement per WO 627385.
- Retest of reactor coolant loop 1 residual heat removal pump suction isolation valve 1-RC-V-22 on April 11, 2011, following overhaul per WO 1196480.
- Retest of "A" train RHR pump 1-RH-P-8A following replacement on May 16, 2011, per WO 40085334.
- Retest of "A" emergency diesel generator on May 9, 2011, following failure of the emergency power sequencer during testing per WO 40082703 (CR1645405).
- Retest of SI check valve SI-V-82 in accordance with OX1401.04 on May 22, 2011, following leak seal repair per WO40086371.

b. Findings

No findings were identified.

1R20 Refueling and Outage Activities (71111.20 - 1 sample)

.1 Refueling Outage OR14

a. Inspection Scope

The inspectors completed one refueling and outage activities inspection sample. The inspectors reviewed the operational, maintenance, and testing activities for the fourteenth refueling outage (OR14) starting on April 1, 2011. The documents reviewed are listed in the Attachment.

Review of Outage Plan
The inspectors reviewed the outage plans to evaluate NextEra's ability to assess and manage the outage risk. The inspectors reviewed the outage risk assessment provided in Engineering Evaluation EE-11-02, "OR14 Outage Schedule Initial Shutdown Risk Review."

Monitoring of Shutdown Activities
The inspectors reviewed activities to shut the plant down in accordance with plant procedures. The inspectors observed completion of various activities specified to place the plant in a cold shutdown condition to assess operator performance, communications, command and control and procedure adherence. The inspectors reviewed operator adherence to TS specified cooldown limits. The inspectors performed inspection tours of plant areas not normally accessible during plant power operations to verify the integrity of structures, piping and supports, and to confirm that systems appeared functional.

Refueling Activities and Reactivity Control
The inspectors verified that refueling activities were performed in accordance with procedures OS1000.09 and RS0721. The inspectors independently verified on a sampling basis that requirements for core alteration were met. The inspectors observed NextEra actions during core alterations to assure core reactivity was controlled. The inspectors observed activities from the control room, the reactor cavity and the spent fuel pool at various times. The inspectors verified that fuel movement was tracked in accordance with the fuel movement schedule. The inspectors verified NextEra action to meet the requirements of TS 3.9 for refueling operations, including the requirements for boron concentration and core monitoring using the source range monitors. The inspectors observed communications and coordination of activities between the control room and the refueling stations while fuel handling activities were in progress. The inspectors verified reactivity was controlled in accordance with the requirements of Technical Specification 3.9.

Control of Outage Risk and Activities
The inspectors reviewed daily shutdown risk assessments during OR14 to verify that NextEra addressed the outage impact on defense-in-depth for the critical safety functions: electrical power availability, inventory control, decay heat removal, reactivity control, and containment. The inspectors reviewed how NextEra provided defense-in-depth for each safety function and implemented the planned contingencies in order to minimize overall risk where redundancy was limited or not available. The inspectors periodically reviewed risk updates accounting for schedule changes and unplanned activities. The inspectors reviewed management controls to manage fatigue by reviewing waiver requests and assessments.

Control of Heavy Loads
The inspectors reviewed NextEra's activities to control the lift of heavy loads in accordance with plant procedures and the commitments to NUREG 0612. The inspectors observed the lift preparations and lift activities to verify adherence to established procedures and controls. The inspectors used an operating experience smart sample as a reference for this review.

Clearance Activities and Configuration Control
The inspectors reviewed a sample of risk significant clearance activities and verified tags were properly hung and/or removed, equipment was appropriately configured per the clearance requirement, and that the clearance did not impact equipment credited to meet the shutdown critical safety functions.

Inventory Control
The inspectors reviewed NextEra actions to establish, monitor and maintain the proper water inventory in the reactor during the outage, and in the reactor and spent fuel pool after flooding the reactor cavity for refueling activities. The inspectors reviewed the plant system flow paths and configurations established for reactor makeup and reactivity control, and verified the configurations were consistent with the outage plan.

Reduced Inventory and Mid-Loop Conditions
The inspectors reviewed NextEra's procedures to implement commitments from Generic Letter 88-17 and confirmed that controls for mid-loop operations were in place. The inspectors verified reactor coolant system instrumentation was installed and configured to provide accurate indication. The inspectors reviewed outage activities that were performed during periods when there was a short time-to-boil to assure adequate controls were in place. Periodically, during the decreased inventory conditions, the inspectors verified that the configurations of the plant systems were in accordance with the commitments. During reduced inventory operations, the inspectors observed NextEra's control of distractions to assure the operator could maintain the specified reactor vessel level.

Foreign Material Exclusion
The inspectors reviewed the implementation of Seabrook procedures for foreign material exclusion control (FME) for the open reactor vessel, reactor cavity and spent fuel pool. The inspectors reviewed NextEra actions to verify that FME issues were documented and resolved.

Electrical Power
The inspectors verified that the status of electrical systems met TS requirements and the outage risk control plan. The inspectors verified that compensatory measures were implemented when electrical power supplies were impacted by outage work activities and that credited backup power supplies were available.

RHR System Monitoring
The inspectors observed spent fuel pool (SFP) and reactor decay heat removal system status and operating parameters to verify that the cooling systems operated properly. The review included periodic review of SFP and reactor cavity level, temperature, and RHR flow. The inspectors reviewed system status to verify the proper system alignment was established for vessel and cavity level measurement.

Containment Control
The inspectors reviewed NextEra activities during the outage to control primary containment closure and integrity, and to prepare the containment for closure prior to plant restart. The inspectors performed tours of all levels in the containment throughout the outage and prior to plant startup per procedure OS1015.18 to review NextEra's cleanup and demobilization controls in areas where work was completed to assure that tools, materials and debris were removed. This review focused on the control of

transient combustibles and the removal of debris that could impact the performance of safety systems.

Monitoring Plant Heat up, Approach to Critical and Startup
The inspectors observed operator performance during the plant startup activities performed between April 28, 2011 and May 26, 2011. The inspection consisted of control room observations, plant tours and a review of the operator logs, plant computer information, and station procedures. The inspectors observed pre-job briefs for key evolutions. The inspectors reviewed the preparations for changes in operating modes. The reactor was taken critical on May 23, 2011 at 03:08 a.m., and completed power ascension to 100% FP on May 26, 2011. The inspectors verified, on a sampling basis, that TS, license conditions, and other requirements for mode changes were met. The inspectors verified RCS integrity throughout the restart process by periodically reviewing RCS leakage calculations and by review of systems that monitor conditions inside the containment.

Problem Identification and Resolution
The inspectors reviewed NextEra actions to identify outage related issues and enter them into the corrective action program. This inspection included a review of the corrective actions for Condition Report 1640003. The inspectors reviewed a sample of the corrective actions to verify they were appropriate to resolve the identified issues.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 7 samples)

a. Inspection Scope

The inspectors completed seven surveillance testing inspection samples. The inspectors observed portions of surveillance testing activities for safety-related systems to verify that the system and components were capable of performing their intended safety function, to verify operational readiness, and to ensure compliance with specified TS and surveillance procedures. The inspectors attended selected pre-evolution briefings, performed system and control room walk downs, observed operators and technicians perform test evolutions, reviewed system parameters, and interviewed the system engineers and field operators. The test data recorded was compared to procedural and TS requirements, and to prior tests to identify any adverse trends. The documents reviewed are listed in the Attachment. The following surveillance activities were reviewed:

- EX 1804.033, Containment Spray System 10 Year Air Flow Test, April 8, 2011 (WO1209232 /1209233).
- OX1426.34, Diesel Generator 1A 18 Month Operability Surveillance, April 26, 2011, May 2, 2011 and May 11, 2011 (WO 40077892).
- OX1413.08, Residual Heat Removal Pump 8A Comprehensive Test (IST), April 18, 2011 (WO 01203773).
- RS1748, Subcritical Physics Testing Using SRWM, May 17, 2011.
- OX1426.32, Diesel Generator 1B 18 Month Operability Surveillance, April 24, 2011

through April 25, 2011 (WO 40076902).
- EX1803.003, Local Leakage Rate Testing of FP-V-588 and FP-V-592 (LLRT), April 1, 2011 (WO01209198).
- EX1803.003, Local Leakage Rate Testing of Penetration X35B, Pressurizer Sample Line (LLRT), April 5, 2011 (WO01209191).

The inspectors reviewed deficiencies related to surveillance testing and verified that the issues were entered into the corrective action program. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS01 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

During the period May 9, 2011 through May 12, 2011, the inspector performed the following activities to verify that NextEra was evaluating, monitoring, and controlling radiological hazards for work performed during the OR-14 refueling outage in locked high radiation areas (LHRA) and other radiological controlled areas. Implementation of these controls was reviewed against the criteria contained in 10 CFR Part 20, Technical Specifications, and NextEra's procedures. The documents reviewed are listed in the Attachment.

Radiological Hazards Control and Work Coverage
The inspector identified work performed in radiological controlled areas and evaluated NextEra's assessment of the radiological hazards. The inspector evaluated the survey maps, exposure control evaluations, electronic dosimeter dose/dose rate alarm set points, and radiation work permits (RWP), associated with these areas, to determine if the exposure controls were acceptable. Specific work activities evaluated included transferring the 8A residual heat removal (RHR) pump into the decay heat vault (RWP 65) and hydrolazing the spent fuel pool (SFP) leak-off lines (RWP 61). For these tasks, the inspector attended the pre-job briefings, reviewed relevant documents, and discussed the job assignments with the workers. Radiation protection technicians were questioned regarding their knowledge of plant radiological conditions for these jobs, and the associated controls.

The inspector reviewed the air sample records for samples taken prior to installing steam generator (SG) nozzle dams, to determine if the samples collected were representative of the breathing air zone and analyzed/recorded in accordance with established procedures. During plant tours, the inspector verified that continuous air monitors were strategically located to assure that potential airborne contamination could be identified in a timely manner and that the monitors were located in low background areas.

The inspector toured accessible radiological controlled areas located in the primary auxiliary building, fuel handling building, decay heat vaults, and waste processing building. With the assistance of a radiation protection technician, independent radiation surveys were performed of selected areas to confirm the accuracy of survey data, and the adequacy of postings.

Additionally the inspector reviewed the RWPs developed for other work performed during OR-14 including installation of temporary shielding and scaffolding. In particular, the inspector reviewed the electronic dosimeter dose/dose rate alarm set points, stated on the RWP, to determine if the set points were consistent with the survey indications and plant policy.

Instructions to Workers

By attending pre-job briefings, the inspector determined that workers, performing radiological significant tasks, were properly informed of electronic dosimeter alarm set points, low dose waiting areas, stay times, and work site radiological conditions. By observing work-in-progress, the inspector determined that stay times were appropriately monitored by supervision to assure no procedural limit was exceeded. Jobs observed included transferring the 8A RHR pump into the decay heat vault and hydrolazing SFP leak off lines.

During plant tours, the inspector determined that locked high radiation areas (LHRA) and a very high radiation area (VHRA) had the appropriate warning signs and were properly secured.

The inspector inventoried the keys to LHRAs to determine if the keys were appropriately controlled, as specified by procedure. The inspector discussed with radiation protection supervision the procedural controls for accessing LHRAs and VHRAs and determined that no changes have been made to reduce the effectiveness and level of worker protection.

Contamination and Radioactive Material Control

During plant tours the inspector confirmed that contaminated materials were properly bagged, surveyed/labeled, and segregated from work areas. The inspector observed workers using contamination monitors to determine if various tools/equipment were potentially contaminated and met criteria for releasing the materials from the RCA.

Radiological Hazards Control and Work Coverage

By observing preparations for installing the 8A RHR pump and for hydrolazing the SFP leakoff lines, the inspector determined that workers wore the appropriate protective equipment, had dosimetry properly located on their bodies, and were under the positive control of radiation protection personnel. Supervisory personnel specified the roles and responsibilities of each worker and reviewed the potential job hazards to assure that exposure was minimized and that industrial safety measures were implemented.

Radiation Worker Performance

During job performance observations, the inspector determined that workers complied with RWP requirements and were aware of radiological conditions at the work site. Additionally, the inspector determined that radiation protection technicians were aware of RWP controls/limits applied to various tasks and provided positive control of workers to reduce the potential of unplanned exposure and personnel contaminations.

Problem Identification and Resolution
A review of Nuclear Oversight field observations (OR-14 Daily Quality Summaries) reports, dose/dose rate alarm reports, personnel contamination event reports and associated condition reports, was performed to determine if identified problems and negative performance trends were entered into the corrective action program and evaluated for resolution and to determine if an observable pattern traceable to a similar cause was evident.

Relevant condition reports (CR), associated with radiation protection control access and radiological hazard assessment, initiated between January and May 2011, were reviewed and discussed with NextEra staff to determine if the follow up activities were being performed in an effective and timely manner, commensurate with their safety significance.

b. Findings

No findings were identified.

2RS02 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

During the period May 9, 2011, through May 12, 2011, the inspector performed the following activities to verify that NextEra was properly implementing operational, engineering, and administrative controls to maintain personnel exposure as low as is reasonably achievable (ALARA) for tasks performed during the OR-14. Implementation of this program was reviewed against the criteria contained in the 10 CFR Part 20, applicable industry standards, and NextEra's procedures. The documents reviewed are listed in the Attachment.

Radiological Work Planning
The inspector reviewed pertinent information regarding site cumulative exposure history, current exposure trends, and the ongoing exposure challenges for the outage. The inspector reviewed various OR-14 ALARA plans.

The inspector reviewed the exposure status for tasks performed during the outage and compared actual exposure with forecasted estimates contained in various project ALARA plans (AP). In particular, the inspector evaluated the effectiveness of ALARA controls for all jobs that were estimated to exceed 5 person-rem. These jobs included reactor vessel disassembly/reassembly (AP 11-01), steam generator (S/G) eddy current testing (ECT) (AP 11-02), and reactor vessel nozzle walk downs (AP 11-13).

The inspector reviewed the ALARA plans and associated Work-In-Progress (W-I-P) ALARA reviews for those jobs whose actual dose approached the forecasted estimate. Included in this review were the W-I-P's for cavity decontamination, reactor coolant pump seal replacement/motor maintenance, and scaffolding installation.

The inspector evaluated the departmental interfaces between radiation protection, operations, maintenance crafts, and engineering to identify missing ALARA program elements and interface problems. The evaluation was accomplished by interviewing site staff, reviewing outage W-I-P reviews, and reviewing recent station radiation safety committee (RSC) meeting minutes. Included in this review were the actions taken by the RSC to lower outage project dose goals, as a result of lowering the plant's source term by an effective primary system cleanup.

Verification of Dose Estimates
The inspector reviewed the assumptions and basis for the OR-14 ALARA forecasted exposure. The inspector also reviewed the revisions made to various outage project dose estimates due to a reduced source term (i.e., lower dose rates); including reactor disassembly/reassembly activities, reactor coolant pump (RCP) maintenance, and steam generator maintenance.

The inspector evaluated the implementation of the NextEra's procedures associated with monitoring and re-evaluating dose estimates and allocations when the forecasted cumulative exposure for tasks exceeded the actual exposure. Included in the review were W-I-P reports, that evaluated the effectiveness of ALARA measures, including source term controls, and actions by the RSC to subsequently lower dose goals from the original estimates.

Additionally, the inspector reviewed the exposures for the ten (10) workers receiving the highest doses for 2011 to confirm that no individual exceeded the regulatory limits or performance indicator thresholds.

Source Term Reduction and Control
The inspector reviewed the status and historical trends for the source term. Through review of survey maps and interviews with the Radiation Protection Manager, the inspector evaluated recent source term measurements and control strategies. Specific strategies being employed included use of macro-porous clean up resin, use of submersible ion exchange filters in the reactor cavity, and installation of permanent/temporary shielding.

The inspector reviewed the effectiveness of temporary shielding by reviewing pre/post-installation radiation surveys for selected components having elevated dose rates. Shielding packages reviewed included those placed on the RHR piping, pressurizer spray piping, steam generator penetrations, and RCP piping.

Job Site Inspections
During plant tours, the inspector assessed the implementation of ALARA controls specified in APs and RWPs, performed during OR-14. These activities include work on the 8A RHR pump (AP 11-019) and hydrolazing SFP leak off lines. Workers were questioned regarding their knowledge of job site radiological conditions and ALARA measures applied to their tasks.

Problem Identification and Resolution
The inspector reviewed elements of NextEra's corrective action program related to implementing the ALARA program to determine if problems were being entered into the program for timely resolution, the comprehensiveness of the cause evaluation, and the effectiveness of the corrective actions. Specifically, recent condition reports related to programmatic dose challenges, personnel contaminations, dose/dose rate alarms, and the effectiveness in predicting and controlling worker exposure were reviewed.

b. Findings

No findings were identified.

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

a. Inspection Scope

During the period May 9, 2011 through May 12, 2011, the inspector performed the following activities to verify that in-plant airborne concentrations of radioactive materials are being controlled and monitored, and to verify that respiratory protection devices are properly selected and used by qualified personnel. Implementation of these programs was evaluated against the criteria contained in 10 CFR Part 20, applicable industry standards, and NextEra's procedures. The documents reviewed are listed in the Attachment.

Engineering Controls
The inspector evaluated the use of portable HEPA ventilation systems installed in various plant areas during the OR-14 outage. The inspector determined that the ventilation systems were located at work locations; e.g., steam generators, and the 8A RHR pump cubicle where airborne contamination could potentially occur. The inspector reviewed testing records for portable HEPA ventilation systems to determine that procedural performance criteria were met.

Respiratory Protection
The inspector reviewed the use of respiratory protection devices worn by workers. The inspector reviewed initial radiation survey and air sampling records for S/G nozzle dam installations in the A through D hot and cold legs, associated RWPs, and APs to determine if the use of respiratory protection devices was commensurate with the potential external dose that may be received when wearing these devices. Additionally, the inspector evaluated the use of respiratory protection; i.e., Delta Suits, for other outage tasks, including cavity decontamination.

Problem Identification and Resolution
The inspector reviewed elements of NextEra's corrective action program related to implementing the airborne monitoring program to determine if problems were being entered into the program for timely resolution, the comprehensiveness of the cause evaluation, and the effectiveness of the corrective actions. Specifically, condition reports related to monitoring challenges, personnel contaminations, dose assessments, and the reliability of monitoring equipment were reviewed.

b. Findings

No findings were identified.

2RS04 Occupational Dose Assessment (71124.04)

a. Inspection Scope

During the period May 9, 2011 through May 12, 2011, the inspector performed the following activities to verify the accuracy and operability of personal monitoring equipment and the effectiveness in determining a worker's total effective dose equivalent. Implementation of these programs was evaluated against the criteria contained in 10 CFR Part 20, applicable industry standards, and NextEra's procedures. The documents reviewed are listed in the Attachment.

External Dosimetry
The inspector verified that NextEra's dosimetry processor was accredited by the National Voluntary Laboratory Accreditation Program (NVLAP). The inspector verified that the approved dosimeter irradiation categories were consistent with the types and energies of the site's source term. The inspector reviewed NextEra's semi-annual quality control evaluation; i.e., TLD blind spiking, of the dosimetry processor.

The inspector confirmed that NextEra has developed "correction factors" to address the response differences of electronic dosimeters as compared to thermoluminescent dosimeters.

Internal Dosimetry
The inspector evaluated the equipment and methods used to assess worker dose resulting from the uptake of radioactive materials. Included in this review were bioassay procedures, whole body counting equipment (FastScan, Chair counter, portal contamination monitors) calibration checks and operating procedures, and the analytical results for 10 CFR Part 61 samples.

The inspector determined that the procedural methods include techniques to distinguish internally deposited radioisotopes from external contamination, methods to assess dose from hard-to-measure radioisotopes, and methods to distinguish ingestion pathways from inhalation pathways.

The inspector reviewed the results from two whole body counts to assess the adequacy of the counting time, background radiation contribution, and the nuclide library used for assessing deposition. No individual exposure exceeded a committed effective dose equivalent (CEDE) of 10 mrem.

Special Dosimetric Situations

Declared Pregnant Workers
The inspector reviewed the procedural controls, and associated records, for managing declared pregnant workers (DPW) and determined that no DPWs were employed during the outage. The inspector reviewed the procedural controls to assure compliance with 10 CFR Part 20.

Multi-Dosimetry Methods
The inspector reviewed NextEra's procedures for monitoring external dose where significant dose gradients exist at the work site. For OR-14, external effective dose

equivalent (EDEX) methods were used to evaluate personnel exposure for installing/removing steam generator nozzle dams. The inspector reviewed the dosimetric results for these jobs. The inspector confirmed that in addition to the TLDs worn, workers also wore electronic dosimeters, equipped with telemetry, to assure that dose fields were promptly monitored by radiation protection technicians.

Problem Identification and Resolution
The inspector reviewed elements of NextEra's corrective action program related to implementing the dosimetry program to determine if problems were being entered into the program for timely resolution, the comprehensiveness of the cause evaluation, and the effectiveness of the corrective actions. Specifically, condition reports related to dose assessments, personnel contaminations, and dose/dose rate alarms were reviewed.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152 – 2 sample)

.1 Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As specified by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the Seabrook corrective action program (CAP). This review was accomplished by accessing NextEra's computerized database. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

As specified by Inspection Procedure 71152, "Problem Identification and Resolution," the inspectors performed a semi-annual review of site issues to identify trends that might indicate the existence of more significant safety issues. The inspection included a review of repetitive or closely-related issues documented by NextEra outside of the corrective action program, such as assessment reports, trend reports, performance indicators, major equipment problem lists, system health reports, and maintenance or corrective action program backlogs. The inspectors reviewed the Seabrook corrective action program database for the first and second quarters of 2011, to assess CRs written in various subject areas (equipment problems, human performance issues, etc.), as well as individual issues identified during the NRCs daily CR review (Section 4OA2.1). The inspectors reviewed the 2011 First Quarter trend reports by the operations, security and nuclear projects departments, together with the Fourth Quarter 2010 trend report to verify that NextEra was appropriately evaluating and trending adverse conditions in accordance with procedure PI-AA-207, "Trend Coding and Analysis."

b. Assessment and Observations

No findings were identified. The inspectors did not identify any trends that NextEra had not identified. The inspectors reviewed a sample of issues and events that occurred over the past two quarters that were documented in the corrective action program. The inspectors verified that NextEra appropriately considered identified issues as emerging trends, and in some cases, verified the adequacy of the actions completed or planned to address the identified trends.

NextEra noted the need for continued focus on human performance. NextEra completed a common cause evaluation for an adverse trend in human performance in Operations (CR594198) with improvements noted in the first quarter of 2011. During periodic meetings with station management, the inspectors discussed NRC observations related to human performance. One example included the inadvertent loss of 345KV Line 394 (CR 1640003) that was caused by a combination of inadequate work package instructions and inadequate worker knowledge of tagout conditions. Another example included the inadequate performance of a reactor coolant system (RCS) leakage surveillance per Technical Specification 4.6.2.1.e (CR1663219), in which valve RC-V147, whose position is indicated on the main control board, remained closed for thirty (30) days. While the procedures used for the RCS leakage surveillance could be enhanced, the cause of the issue was the failure to use fundamental operator skills during the performance of routine duties. NextEra corrective actions include a renewed emphasis on operator fundamentals in the operator training program. NextEra continues to address human performance site wide through procedure enhancements, management observations and a focus on procedure compliance in continuing training sessions.

NextEra continued to focus on equipment performance and reliability. Performance problems with secondary plant equipment continue to challenge operators and have resulted in the need to reduce plant power or take the turbine offline three times in three quarters (CRs 591828,1616988,1657622), as reflected in an adverse trend in the NRC Performance indicator for Unplanned Power changes. During periodic meetings with station management, the inspectors discussed emergent equipment issues that impacted safety system availability [e.g., service water system corrosion (CR1633034), EDG sequencer failure (CR1645405), A RHR pump seal leakage (CR1647943)] or impacted the primary system boundary [e.g., SI check valve leakage (CR1652573) and safety valve RC-V117 leakage (CR1662418). NextEra continues to use the preventive maintenance optimization process and the plant health committee reviews of system health reports to focus on equipment issues. Self-assessments have been effective to identify the need for additional actions to address service water system piping degradation (CR1637922).

4OA5 Other Activities

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

The inspectors assessed the activities and actions taken by NextEra to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. This included (i) an assessment of NextEra's capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis on strategies related to the spent fuel pool, as required by NRC Security Order Section B.5.b issued February 25, 2002, as committed to in severe accident management guidelines, and as specified by 10 CFR 50.54(hh); (ii) an assessment of NextEra's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63 and station design bases; (iii) an assessment of NextEra's capability to mitigate internal and external flooding events, as specified by station design bases; and (iv) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flood events, which were performed by NextEra to identify any potential loss of function of this equipment during seismic events possible for the site. Inspection Report 05000443/2011009 (ML111300174) documented detailed results of this inspection activity.

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

On May 20, 2011, the inspectors completed a review of NextEra's severe accident management guidelines (SAMG), implemented as a voluntary industry initiative in the 1990's, to determine (i) whether the SAMGs were available and updated, (ii) whether NextEra had procedures and processes in place to control and update its SAMGs, (iii) the nature and extent of NextEra's training of personnel on the use of SAMGs, and (iv) licensee personnel's familiarity with SAMG implementation. The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant-specific results for Seabrook Station were provided in an Attachment to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated May 27, 2011 (ML111470361).

4OA6 Meetings, Including Exit

On July 13, 2011, the resident inspectors presented the results of the second quarter routine integrated inspections to Mr. E. Metcalf and Seabrook Station staff. The inspectors also confirmed with NextEra that no proprietary information was reviewed by inspectors during the course of the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

NextEra Personnel

J. Ball, Maintenance Rule Coordinator
K. Boehl, Health Physics Analyst
B. Brown, Supervisor, Civil Engineering
V. Brown, Senior Licensing Analyst
K. Browne, Operations Manager
M. Collins, Manager, Design Engineering
W. Cox, Radiological Engineer
R. Gutherie, Plant System Engineer
F. Haniffy, Senior Radiation Protection Analyst
L. Hansen, Plant Engineering
N. Levesque, Plant Engineering
E. Metcalf, Plant General Manager
W. Meyer, Radiation Protection Manager
M. O'Keefe, Licensing Manager
M. Nadeau, System Engineer, Control Building Air Handling
D. Perkins, Supervisor, Radiation Protection Technical Services
M. Scannell, Radiation Protection Technical Specialist
R. Sterritt, ALARA Coordinator
T. Vassallo, Principal Engineer - Nuclear
J. Walsh, Nuclear Steam Supply System, Supervisor

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened:

05000443/2011003-02	URI	Inadequate 50.59 Screening for Design Change EC 272057
05000443/2011003-03	URI	Operability Evaluation for Degraded Concrete in ASR Affected Plant Structures

Opened and Closed:

05000443/2011003-01	NCV	Inadequate Control of Combustible Materials
05000443/2011003-04	NCV	Untimely Operability Determination for Degraded Concrete Structures Housing Safety-Related Equipment
05000443/2011003-05	NCV	Inadequate Operability Determination for Reduced EDG HX Cooling Water Flow

Closed:

05000443/2515/183	TI	Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event (Section 4OA5.1)
05000443/2515/184	TI	Availability and Readiness Inspection of Severe Accident Management Guidelines (Section 4OA5.2)

Discussed:
None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

OP-AA-102-1002, Seasonal Readiness, Revision 0
OAI.42, Operations Department – Severe Weather Plan Implementation
OS1200.03, Severe Weather Conditions, Revision 18
NM11800, Hazardous Condition Response and Recovery Plan
ON1490.09, Summer Readiness surveillance, Revision 5
ON0443.59, Yard Hydrant Semi Annual Inspection, Revision 5
2011 Summer Readiness Site Certification
SBK 11-018, Nuclear Oversight Report – Summer Readiness
Condition Report; AR1655329, 1653764, 1607562
Work Order: 40083993, 40038324, 1384685, 40038436
ODI 61, Redeclaration / Joint Owners & NDDO Notification Guideline, Revision 47
ODI 90, 345kV Electrical Disturbance Communication, Analysis, & Reporting Guideline, Revision 6
OS1246.02, Degraded Vital AC Power, Revision 10
Seasonal Readiness Review – System Engineering
ER1.1, Classification of Emergencies, Revision 49
Operations Department Turnover Report
Daily Status Report
Station Operating Logs - various

Section 1R04: Equipment Alignment

OS 1412.09, Rev. 7, PCCW Monthly Flow Check
OX 1412.05, Rev. 8, Monthly PCCW Loop A Valve Verification
Drawings 1-CC-B20205, 1-CC-B20206, 1-CC-B20207
UFSAR Section 9.2.2 Cooling for Reactor Auxiliaries
Work Orders 40040600, 40073132
OX1416.01, Monthly Service Water Valve Verification
OX1416.06, Service Water Discharge Valves Quarterly Test and 18 Month Position Verification
System Health Report – Service Water System
Operations Logs - various
PID: 1-SW-B20795, 1-SW-B20794, 1-NHY-202476
UFSAR Section 9.2, 7.3
Technical Specifications 3.7.4 Service Water System/Ultimate Heat Sink
Detailed System Text – Service Water System
Plant Engineering Action Plan Register
Operations Logs – various
OS1013.03, Residual Heat Removal System Train A Startup and Operation, Revision 21
OS1013.04, Residual Heat Removal System Train B Startup and Operation, Revision 22
OS1001.11, Reactor Coolant System Shutdown Level, Revision 5
OS1016.03, Service Water Train A Operation, Revision 11
OS1016.04, Service Water Train B Operation, Revision 13
OS1016.05, Service Water Cooling Tower Operation, Revision 19

Section 1R05: Fire Protection

Fire Protection Pre Fire Strategies
Fire Impairment List
Technical Requirement 11 Fire Rated Assemblies
Technical Requirement 12 Fire Detection Instrumentation

UFSAR Section 9.5.1 Fire Protection Systems
UFSAR Section 13.2.2.9 Fire Protection Personnel
OS1200.00A, Fire Hazards Analysis for Affected Area / Zone – Appendix A
OS1200.00, Response to Fire or Fire Alarm Actuation, Revision 15
NUREG 1805 Chapter 8
FP 2.2, Control of Combustible Materials, Revision 13 (draft)
Response to NRC Fire Protection issue 1-SS-CP-166B
Fire Zone W-F-1A, 1B-Z & W-F-5-0
Station Operating Logs - various

Section 1R08: Inservice Inspection

ES1807.002 Rev 9, Liquid Penetrant Examination – Solvent Removable
ES1807.003 Rev 8, Magnetic Particle Examination
ES1807.001 Rev 7 CH 2, Visual Examination Procedure for Welding
ES03-01-27 Rev 2, PDI Generic Procedure for Manual Ultrasonic through Wall and Length Sizing of Ultrasonic Indications in Reactor Pressure Vessel Welds (PDI-UT-7)
ES10-01-32 Rev 00, Remote Inservice Examination of Reactor Vessel Nozzle to Safe End, Nozzle to Pipe, and Safe End to Pipe Welds Using the Nozzle Scanner (PDI-ISI-254-SE-NB, Rev 1)
ES1807.025 Rev 5, Inservice Inspection (ISI) Visual Examination Procedure (VT-2)
ES1807.012 Rev 6, Ultrasonic Thickness Measurements
MA 10.3 Rev 5, Boric Acid Corrosion Control Program
P1-AA-102 Rev 3, Non-Safety Operating Experience Program
P1-AA-102-1001 Rev 4, Operating Experience Program Screening and Responding (Incoming)
AR 00569156, B1 Boric Acid Leak in Outlet Isolation Valve Packing Area
AR 01636221, Medium to Heavy Boric Acid Leakage from RHR Pump Suction Packing
AR 00210637, Boric Acid Leak at Packing on Valve FCV121
AR 00219427, Boric Acid Leak from Transmitter Fitting RC-FT-415
AR 00213435, Boric Acid Leak at Packing Charging Header Vent Valve 1-CS-V-836
AR 01640609, Reactor Vessel Hot Leg Post MSIP Exam (158 degree nozzle)

Examination Reports

1198488, Liquid Penetrant of SW-1814 Joint F0104, dwg SKEC145189-2000
1198488, Magnetic Particle Exam of SW-1814 Joint F0104, dwg SK-EC145189-2000
1198488, Ultrasonic Examination of SW1814-1-156-24 Thickness Report
1-SW-1814-001, VT-2 Visual Examination Form, Service Water System
40055977-01, Magnetic Particle Exam Data Sheet (SW) Weld F0105, 106 and 107
01209165, Visual Examinat5ion (VT-2) of Pressurizer Heater Sleeves
1208874, Remote Visual (VT-2) Examination of RPV Bare Metal Upper Head
SP-SWOL-DS01, Ultrasonic Exam of Pressurizer Spray Nozzle (Phased Array)
S-SWOL-DS01, Ultrasonic Exam of Pressurizer Surge Nozzle (Phased Array)

Work Orders

WO 01199620 01, 1-CS-FCV-121 Overhaul Valve Replace Valve Trim
WO 01202400 01, CS-V-836-B3 (Wet) Boric Acid Leak at Packing
WO 01198488 02, Weld Repair of Salt Service Water Line Install Repair Cap
WO 40055977 01, Fabrication of Salt Service Welded Pipe Replacement Spool Piece

Work Requests

WR 94002854, Boric Acid Leak at Charging Flow Control Valve FCV 121

WR 94003420, Charging Header Vent Valve Boric Acid Leak CS-V-836
WR 94002533, Fabricate and Install Reducer in Line 1814-01 Salt Service Water

Welding Procedures (WPS) and Procedure Qualification Records (PQR)
WPS ES0815.004, Manual Gas tungsten (GTAW) and Shielded Metal Arc (SMAW) Welding of Carbon Steel to Carbon Steel (P1 to P1)
WPS ES0815.004, Manual SMAW of carbon steel to carbon steel PQR SBK1-815.004-1 Weld Procedure Qualification Record GTAW/SMAW of P1 to P1 with Post Weld Heat Treatment (PWHT)
PQR SBK1-815.004-2 WPS for P1 to P1 without PWHT
UC 371 & 391, Welder Performance Qualification Record Review to use ES0815.004-1

Drawings
SK-EC270505-2000, Installation Detail Service Water Piping Repair (SW 1814)
SK-EC270505-2001, Fabrication Detail Service Water Piping Repair (SW 1814)

Miscellaneous
AR 220564, Self Assessment – Boric Acid Corrosion Control Program
2010 3rd Qtr, Program Health Report - Boric Acid Corrosion Control Program
2010 4th Qtr, Program Health Report – Boric Acid Corrosion Control Program
CR 05-11634, Engineering Evaluation for 1-CS-FCV-121
CR01636130, UT results of SW Piping Indicates Wall Thinning
CR (AR 00213435), Boric Acid Corrosion Evaluation (EDI 30560) Valve 1-CS-V-836
CR (AR 210637), Boric Acid Corrosion Control ASME Bolting Evaluation 4-10-2011
MSE#:05-040, Maintenance Support Eval for Valves CS-FCV-121 and 1-CS-HCV-182
EC145189, ASME XI Repair/Replacement Plan Traveler Component SW-1814
EC 271779 R0, Temp Installation for Repair of Section of SW-1814-001
EDI 30560, Boric Acid Corrosion Evaluation of Valve 1-CS-V-836 Vent Valve
SIIR, Inservice Inspection Program Plan for 3rd Ten Year Interval

Section 1R11: Licensed Operator Requalification Program
OS1000.02, Plant Startup from Hot Standby to Minimum Load, Revision 20
OS1000.05, Power Increase, Revision 16
OS1000.07, Approach to Critical, Revision 10
OS1007.01, Automatic and Manual Rod Control, Revision 10
OS1056.03, Containment Penetrations, Revision 6
OS1213.01, Loss of RHR While in Reduced Inventory, Revision
ON1031.02, Starting and Phasing the Turbine Generator, Revision 26
ON1031.13, Post Maintenance Turbine Startup, Revision 12
RS1735, Reactivity Calculations, Revision 4
ODI.101, Guarded Equipment Recommendations for Refueling Outages, Revision 5
ODI.82, Mode Change Notice, Revision 15

Section 1R12: Maintenance Effectiveness
System Health Report – RHR system
Maintenance Rule Performance and Scope Report
UFSAR Section
Condition Reports 1612061, 1632409, 1633034, 1636533
PODs for CR1612061/1632409/1633034
Drawing 1-SW-B20795
OR14 Service Water Inspections / Results

OR14 Service Water Piping Assessment
AR1939781 – SW Train B Pump House Inspection
Work Orders 40080265, 0062557102, 40078949
EC272058, SW Pipe Repairs
Design Engineering Review for Service Water Pipe SW-1814 83 Day UT Results
AR1637922 – DQS of Service Water Corrective Actions
System Health Reports – Service Water System
Plant Engineering Action Register
Condition Reports 2010-2011
Work Requests 2010-2011
Station Operating Logs - various

Section 1R13: Maintenance Risk and Emergent Work

OR14 Outage Schedule Initial Shutdown Risk Review Rev. 0
OR14 SW Extent of Condition Inspection Matrix 4/13/2011
WM-AA-1000 Work Activity Risk Management Rev. 6
OS 1016.11 Contingency Ocean Pump Restoration for SW Work Activities with Ocean Service Water Pumps not in Service. Revision 01
UFSAR 9.2.5 Ultimate Heat Sink
Drawings 1-SW-B20795, 1-SW-B-B20794
M-Rule a(4), Risk Assessment Reports
Station Operating Logs - various
AR 1640932, 1610327, 1639921, 1631769, 1631776, 1640932, 1640932
WO 1199040, 1205038, 1203446, 249348, 00626035
Lift Plan and Rigging Evaluation – A RHR Pump Roof Plug and RHR motor
TS – Various
RHR leak rate summary
Plant Engineering Register
EX1801.002, Leakage Reduction Program Surveillance, Revision 9
MS0523.24, Ingersoll-Rand Residual Heat Removal Pump Maintenance, Revision 7
OS1213.02, Loss of RHR while Operating at Reduced Inventory or Midloop Conditions, Revision 12
OS1215.05, Loss of Refueling Cavity Water, Revision 15
OS1213.01, Loss of RHR During Shutdown Cooling, Revision 15
OS1056.03, Containment Penetrations, Revision 6
ODI.103, Conduct of Infrequently Performed Tests or Evolutions, Revision 0
ODI.101, Guarded Equipment Recommendations for Refueling Outages, Revision 5
Work Orders 40086371 Tasks 1, 2, 3 and 4, WO 1382815
Adverse Condition Monitoring Plan for SI-V82 dated 6/8/11
SI-V-82 Operational Decision Making
MS0526.09, On Stream Leak Repairs, Revision 4
Insulation Removal Evaluation for SI-V-82
IN93-90, Unisolatable Reactor Coolant System Leak Following Repeated Applications of Leak Sealant

Section 1R15: Operability Evaluations

ODM, Operational Decision Making for RC-V-117 Leak (AR 1633034)
Station Operating Logs - various
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LIST OF ACRONYMS

ACI	American Concrete Institute
ADAMS	Agency-wide Documents Access and Management System
ALARA	As Low As is Reasonably Achievable
AMS	Airborne Monitoring System
AP	ALARA Plans
AR	Action Request
ASME	American Society of Mechanical Engineers
ASR	Alkali-silica Reaction
BACC	Boric Acid Corrosion Control (Program)
CAP	Corrective Action Program
CB/ET	Control Building/Electric Tunnel
CEB	Containment Enclosure Building
CEDE	Committed Effective Dose Equivalent
CR	Condition Report
DG	Diesel Generator
DPW	Declared Pregnant Workers
ECT	Eddy Current Testing
EDEX	External Effective Dose Equivalent
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
FBL	Fire Brigade Leader
FHB	Fuel Handling Building
FPP	Fire Protection Program
GTAW	Gas Tungsten Arc Welding
HEPA	High Efficiency Particulate Air
IMC	Inspection Manual Chapter
IP	Inspection Procedure
ISI	In-service Inspection
LHRA	Locked High Radiation Areas
MR	Maintenance Rule
MSIP	Mechanical Stress Improvement Process
MT	Magnetic Particle Test
NCV	Non-cited Violation
NDE	Non-Destructive Examination
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
OR	Outage for Refueling
OD	Operability Deficiency
ODs	Operability Determinations
OM	Operations Management
PAB	Primary Auxiliary Building
PARS	Publicly Available Records
PCCW	Primary Component Cooling Water
PDI	Performance Demonstration Initiative
PMT	Post-maintenance Testing
POD	Prompt Operability Determination

PQR	Procedure Qualification Record
PT	Penetrant Test
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RSC	Radiation Safety Committee
RWP	Radiation Work Permit
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SG	Steam Generator
SM	Shift Manager
SMAW	Shielded Metal Arc Welding
SPF	Spent Fuel Pool
SPM	Scintillation Portal Monitor
SRWM	Subcritical Rod Worth Measurement
SSC	Structures, Systems or Components
SW	Service Water
SWP	Service Water Pump
TI	Temporary Instruction
TLD	Thermoluminescent Dosimeter
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
VHRA	Very High Radiation Area
VT	Visual Test
W-I-P	Work-In-Progress
WO	Work Order
WPS	Weld Procedure Specification
WR	Work Request